FOSTER WHEELER ENVIRONMENTAL CORPORATION

DON ROGERS
Executive Vice President
and Chief Operating Officer

TEL 973-630-8063
FAX 973-630-8526

FW-NRC-ISF-01-0608
November 19, 2001

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

**SUBJECT: IDAHO SPENT FUEL (ISF) FACILITY
FOSTER WHEELER ENVIRONMENTAL CORPORATION
APPLICATION FOR 10 CFR PART 72 LICENSE
DOCKET NO. 72-25**

Dear Sir or Madam:

In accordance with Nuclear Regulatory Commission (NRC) regulations, including 10 CFR Part 72, the Foster Wheeler Environmental Corporation (FWENC) hereby applies for a specific license, and all other necessary licenses and approvals, to receive, transfer, package, and store certain reactor spent fuel and radioactive material associated with spent fuel storage in an independent spent fuel storage installation (ISFSI) to be constructed at the Department of Energy (DOE) Idaho National Engineering and Environmental Laboratory (INEEL). This facility, to be known as the Idaho Spent Fuel (ISF) Facility, represents a key element of the DOE's National Spent Fuel Program for stabilizing its inventory of spent fuel prior to shipment to a permanent repository.

The ISF Facility will be used to store spent fuel and associated radioactive material from (1) the first and second cores of the Peach Bottom 1 reactor, (2) the Shippingport reactor, and (3) certain TRIGA reactors. Once the ISF Facility is operational, the DOE will transport the spent fuel from storage locations on the INEEL to the ISF Facility. The fuel transfer will occur completely within the boundaries of the INEEL site and will be conducted in accordance with INEEL procedures and DOE orders.

The ISF Facility license application contains information required by 10 CFR Part 72 and other applicable NRC regulations, and has been prepared considering the guidance provided in applicable Regulatory Guides, Standard Review Plans, and Interim Staff Guidance documents. This application demonstrates that the FWENC ISF Facility complies with the standards and requirements of the Atomic Energy Act of 1954, as amended, and the regulations of the NRC, including Part 72, particularly 10 CFR 72.40, *Issuance of License*. Provided under separate cover is a cross-reference matrix that identifies the location of information in the license application to the acceptance criteria of NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*.

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This submittal consists of the following parts:

1. License Application, with the following documents included as appendices:
   a. A request for exemption from seismic design requirement,
   b. A training program as required by 10 CFR 72.192, *Operator Training and Certification Plan.*
   c. A decommissioning plan as required by 10 CFR 72.30, *Proposed Decommissioning Plan.*
   d. Proposed Technical Specifications required by 10 CFR 72.26 and 72.44, *Proposed Technical Specifications.*
2. A Safety Analysis Report (SAR) as required by 10 CFR 72.24, *Safety Analysis Report.*
3. An emergency plan as required by 10 CFR 72.32, *Emergency Plan.*
4. An environmental report as required by 10 CFR 72.34, *Environmental Report.*
5. Security information as required by 10 CFR 72, Subpart H, *Physical Protection Plan*, which is being forwarded under separate cover with a request that it be withheld from public disclosure.

The Quality Program Plan for Foster Wheeler Environmental Corporation's Idaho Spent Fuel Facility was previously submitted to the NRC, under separate cover, on March 31, 2001.

It also should be noted that this application was developed consistent with both the NRC regulations and the DOE INEEL site procedures for emergency response and safeguards protection. FWENC remains alert to developments that may arise from considerations following the events of September 11, 2001.

FWENC also is requesting, as part of this application, an exemption from the seismic design requirements of 10 CFR 72.102 to facilitate use of a probabilistic seismic hazard analysis for seismic risk commensurate with an ISFSI. The NRC regulation for seismic design requirements for ISFSI, Section 72.102, currently requires use of a deterministic seismic hazard analysis in accordance with 10 CFR Part 100, Appendix A to determine the design basis earthquake for an ISFSI. This request for exemption is consistent with the exemption granted by NRC for the nearby TMI-2 ISFSI, and with the developments at this time in the current NRC rulemaking on this issue. The specific request for the exemption is presented as Appendix A to the License Application.

FWENC believes this application is responsive to regulatory requirements and guidelines. It should be noted that some information in the application is marked as being for later submittal to the NRC due to delays in receipt from the DOE. The information to be provided later generally concerns the casks that the DOE will use to transport the spent nuclear fuel to the ISF Facility. FWENC expects that current application information, assumptions and calculations will bound the anticipated supplementary DOE information. The DOE commitment and schedule for providing this information is included in this application. Such information will be submitted, as appropriate, as it is received and reviewed. Please refer to Appendix A of the Safety Analysis

Report for specific information in this regard. The absence of this supplementary information does not adversely impact the sufficiency of this application for purposes of docketing, nor should it impact the ability of the NRC to review this application at this time. Of course, if FWENC were to receive additional information from the DOE which modifies any statements made in the application, then FWENC would promptly notify the NRC in accordance with regulatory requirements.

FWENC requests prompt NRC acceptance review of this application for docketing. In that regard, discussions with the NRC have been initiated regarding a meeting with your staff and consultants to address key application provisions. FWENC is prepared and welcomes the opportunity for an early public meeting with the NRC reviewers to facilitate timely review of the application. As discussed with NRC staff, this meeting has been scheduled for December 11, 2001 in San Antonio, TX. Additionally, as reviewers become more familiar with the application contents, a follow-up meeting to further assist review of this application for docketing may be appropriate.

Construction of the ISF Facility is scheduled to begin in July 2003. Operations are scheduled to commence in June 2005. In order to support this schedule, NRC issuance of the ISF Facility Materials License is respectfully requested by May 2003.

Consistent with conversations with James R. Hall, the Spent Fuel Project Office Project Manager, thirteen copies of each portion of the license application are being delivered directly to him. The original and two remaining required copies of each portion of the license application are provided herein. As stated previously, the ISF Physical Protection Plan is being forwarded under separate cover. If you have any questions, please contact me at (973) 630-8063 or Ronald Izatt, ISF Facility Project Manager, at (509) 372-5808.

Sincerely,

Donald I. Rogers, Jr.
Executive Vice President and
Chief Operating Officer
Foster Wheeler Environmental Corporation

JS/naw
Enclosures (one original and two copies):

1. License Application
2. Safety Analysis Report
3. Emergency Plan
4. Environmental Report

cc: James R. Hall, SFPO Project Manager (NRC) (Letter with thirteen copies)
Ellis W. Merschoff, Region IV Administrator (NRC) (Letter only)

FOSTER WHEELER ENVIRONMENTAL CORPORATION

STATE OF NEW JERSEY)
)
)
COUNTY OF MORRIS)

Donald I. Rogers, Jr. being duly sworn, deposes and states that he is the Executive Vice President and Chief Operating Officer, Foster Wheeler Environmental Corporation, the licensee applicant, herein; that the statements made in this document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute and file this document on behalf of said applicant.

Donald I Rogers, Jr.
Executive Vice President and Chief Operating Officer
Foster Wheeler Environmental Corporation

On this day personally appeared before me, Donald I. Rogers, Jr., to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act.

GIVEN under my hand and seal this 16th. day of November, 2001.

Caroline A. Florio

Notary Public in and for the State of New Jersey

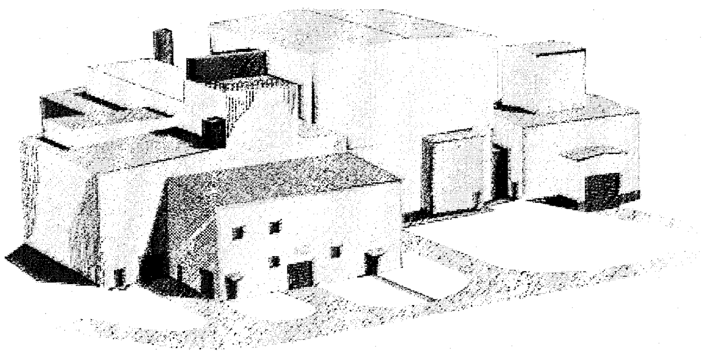
ISF FORM: ISF-L-0002-2, Rev. 1, Effective Date: 09/18/01

Caroline A. Florio
Notary Public State of New Jersey
Commission Expires 10/21/2004

License Application

Idaho Spent Fuel Facility

Docket No. 72-25

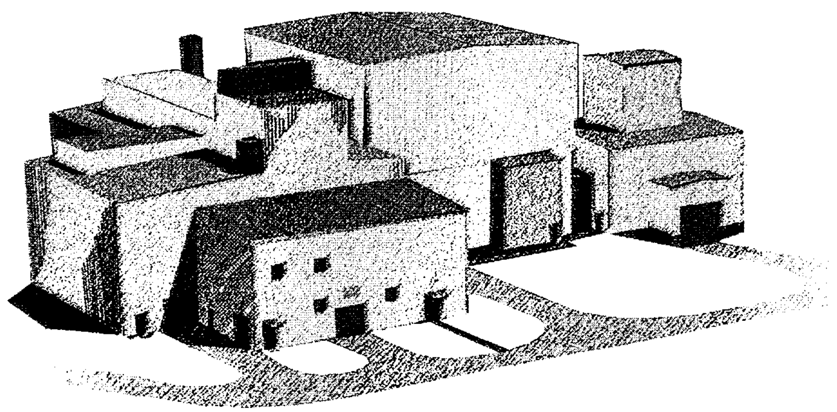


FOSTER WHEELER ENVIRONMENTAL CORPORATION

License Application

Idaho Spent Fuel Facility

Docket No. 72-25



ISF-FW-RPT-0127

FOSTER WHEELER ENVIRONMENTAL CORPORATION

Contents

Appendices

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APPLICATION FOR LICENSE TO CONSTRUCT AND OPERATE AN INDEPENDENT SPENT FUEL STORAGE INSTALLATION

1.0 GENERAL AND FINANCIAL INFORMATION

1.1 APPLICATION FOR LICENSE

In accordance with the requirements of Title 10, Code of Federal Regulations, Part 72 (10 CFR 72), Foster Wheeler Environmental Corporation (FWENC) hereby submits this License Application to construct and operate an Independent Spent Fuel Storage Installation (ISFSI) at the site of the Idaho National Engineering and Environmental Laboratory (INEEL) located in Butte County, Idaho (Ref. 1). The proposed facility is named the Idaho Spent Fuel (ISF) Facility.

This application for the proposed ISFSI contains information required by the provisions of 10 CFR 72, Subpart B and was prepared using the guidance provided in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.50, *Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste* (Ref. 2). This application consists of the following:

a) License Application, ISF-FW-RPT-0127.

b) Technical information and Safety Analysis Report (SAR) required by 10 CFR 72.24 provided as a separate document titled, *Safety Analysis Report*, ISF-FW-RPT-0033.

c) A request for specific exemption from the seismic design requirements of 10 CFR 72.102(f)(1), provided as an appendix to this License Application.

d) A training program as required by 10 CFR 72.192 provided as an appendix to this License Application, *Operator Training and Certification Plan*, ISF-FW-PLN-0031.

e) Security information as required by 10 CFR 72, Subpart H, provided as a separate document titled, *Physical Protection Plan,* ISF-FW-PLN-0029. This is being submitted under separate cover.

f) A decommissioning plan as required by 10 CFR 72.30 provided as an appendix to this License Application, *Proposed Decommissioning Plan,* ISF-FW-PLN-0027.

g) An emergency plan as required by 10 CFR 72.32 provided as a separate document titled, *Emergency Plan*, ISF-FW-PLN-0021.

h) An environmental report as required by 10 CFR 72.34 provided as a separate document titled, *Environmental Report*, ISF-FW-RPT-0032.

i) Proposed technical specifications required by 10 CFR 72.26, *Contents of Application: Technical Specifications*, and 10 CFR 72.44, *License Conditions*, provided as an appendix to this License Application, *Proposed Technical Specifications*, ISF-FW-RPT-0034.

FWENC has previously submitted a description of the FWENC quality assurance program as required by 10 CFR 72.24(n) via letter dated March 31, 2001 (Ref. 3). The *Quality Program Plan*, ISF-FW-PLN-0017, describes the quality assurance program that directs quality-related activities for the ISF Facility.

1.2 APPLICANT

The name of the applicant is Foster Wheeler Environmental Corporation.

The principal address is:

Foster Wheeler Environmental Corporation
1000 The American Road
Morris Plains, New Jersey 07950
973-630-8000 (Telephone)
973-630-8025 (Facsimile)

1.3 DESCRIPTION OF BUSINESS OF APPLICANT

FWENC is a leading environmental consulting, engineering, and waste management firm employing more than 2200 dedicated professionals in 25 U.S. offices and project sites and 19 international locations. FWENC is an indirect wholly owned subsidiary of Foster Wheeler Ltd., a Fortune 500 company in business for more than 100 years. Foster Wheeler has provided services to the U.S. Department of Energy (DOE) and its predecessor agencies for the past 40 years. Foster Wheeler Ltd. is ISO 9001 certified and FWENC is the first U.S. environmental firm to be ISO 14001 certified for all offices and project sites. *Engineering News-Record* consistently ranks Foster Wheeler among the top five environmental contractors in the nation.

Foster Wheeler Ltd. is an international organization that provides engineering services and products to a broad range of industries, including the petroleum and gas, petrochemical, pharmaceutical, chemical processing, and power-generation industries. These services include design, engineering, construction, and procurement, as well as project management, research, plant operation, and environmental services. The company supports the global marketplace with engineering centers strategically positioned around the world. Foster Wheeler Ltd., has more than 12,000 employees worldwide and had annual revenues of approximately $4.0 billion in both 1999 and 2000.

1.4 LEGAL STATUS AND ORGANIZATION

Foster Wheeler Environmental Corporation (FWENC) is an indirect wholly owned subsidiary of Foster Wheeler Ltd. FWENC is a corporation organized and existing under the laws of the State of Texas with its principal office located in Morris Plains, New Jersey. The address of FWENC is provided in Section 1.2. FWENC submits this application on its own behalf and is not acting as an agent or representative of any other person or organization.

Foster Wheeler Corporation Summary Annual Report for 2000 provides an overview of the company and its subsidiaries, its officers and board of directors, and financial highlights. This report can be obtained from the Foster Wheeler website at http://www.fwc.com. Additional information regarding Foster Wheeler Ltd. and the company's reorganization in 2001 can be found in the U.S. Securities and Exchange Commission Form 10-K filings dated March 2001 and May 2001 and Form 10-Q filings dated August 2001.

The current principal officers of FWENC are listed below. All are citizens of the United States. These individuals may be contacted at the principal address for FWENC provided in Section 1.2.

Officer	Title
Sam W. Box	Chairman, President and CEO
Donald I. Rogers, Jr.	Executive Vice President and COO
Martin S. Brown	Executive Vice President
Larry D. Carter	Senior Vice President and Assistant Secretary
Thomas DelMastro	Vice President, CFO, Treasurer and Assistant Secretary
John H. DeFeis, Jr.	Vice President
Henry E. McGuire Jr.	Vice President
Frank G. Teague	Vice President

The current directors of FWENC and their citizenship are presented below. Addresses for these individuals can be obtained by contacting FWENC at the principal address provided in Section 1.2.

Director	Citizenship
John C. Blythe	United Kingdom
Sam W. Box	United States
Martin S. Brown	United States
John S. Lambert	United States
Donald I. Rogers, Jr.	United States

1.5 FINANCIAL QUALIFICATIONS

The ISF Facility will be constructed at the INEEL in Butte County, Idaho. Construction of the ISF Facility is scheduled to begin in July 2003. Operations are scheduled to commence in June 2005.

FWENC will design, construct, and initially operate the ISF Facility under contract with the DOE (Ref. 4). The DOE maintains a current copy of this contract on the DOE Idaho Operations Office website. The contract is available for viewing at http://www.id.doe.gov/doeid/psd/SNFDSPContract.htm. In accordance with the contract, the DOE will make an initial payment to FWENC based on achievement of a specific milestone prior to start of construction. FWENC will be responsible for funding the construction and initial operation of the ISF Facility. FWENC estimates that the construction costs associated with the ISF Facility will be approximately $114 million, based on 1999 dollars. FWENC's parent company, Foster Wheeler Ltd., will provide the necessary interim funding of these activities, pending milestone payments to FWENC by the DOE.

Once the ISF Facility is operational, the DOE will make payments to FWENC on an amortized basis during the transfer and storage of the first 800 fuel handling units (FHUs)[1] of Spent Nuclear Fuel (SNF). These amortized capital costs total approximately $114 million. In addition to the amortizing payments, the DOE will also make payments for the transfer and storage of the remaining SNF at specific unit prices for each SNF type. The total payments inclusive of all fuel types are approximately $31 million.

1 A fuel handling unit (FHU) is a contractually defined term used to provide for the accounting of partial fuel elements. For intact SNF, one FHU is equal to one fuel element.

Also, in accordance with the contract, post storage operation and maintenance of the facility is optional. The DOE has the contractual option (pending necessary license transfer) to assume responsibility for the facility after the initial fuel handling, packaging, and storage operations. Should DOE desire that FWENC continue as the licensee during the post storage operations phase of the project, the contract provides for DOE payments to FWENC of approximately $1.85 million per year.

Furthermore, in accordance with the contract, the DOE retains the ownership of the SNF and remains financially responsible for the eventual decontamination and decommissioning of the ISF Facility. However, in support of this license application, FWENC has prepared a *Proposed Decommissioning Plan* that presents the estimated cost, funding methodology, and plans for ensuring availability of funds associated with the decommissioning of the ISF Facility. Decommissioning is forecast to occur in 2018 at an estimated cost of $23 million, based on 2001 dollars.

1.6 COMMUNICATIONS

The key individuals responsible for the preparation of this license application are:

Ronald D. Izatt	ISF Facility Manager
Randy J. Roberts	Chief Design Engineer
James C. Saldarini	Licensing Manager

It is requested that communications pertaining to this application be sent to:

Ronald D. Izatt, ISF Facility Manager
Foster Wheeler Environmental Corporation
3200 George Washington Way, Suite G
Richland, Washington 99352
(509) 372-5808 (Telephone)
(509) 372-5801 (Facsimile)

2.0 TECHNICAL QUALIFICATIONS

FWENC has overall responsibility for design, licensing, construction, and operation of the ISF Facility. The past experience gained by FWENC during construction and operation of similar nuclear facilities will be applied to the construction and operation of the ISF Facility. In addition, FWENC has engaged key subcontractors with additional specific technical expertise and experience. These subcontractors include: RWE NUKEM, Ltd. (NUKEM), ALSTEC Ltd. (ALSTEC), and Utility Engineering (UE).

NUKEM , formerly AEA, provides key design experience in the remote handling of SNF. Their experience in designing, licensing and constructing the Intermediate Level Waste Facility at Harwell, UK, is directly relevant to this project. ALSTEC, formerly ALSTOM, provides expertise and experience related to the design of SNF storage systems. ALSTEC brings over 30 years of relevant experience in developing fuel storage facilities and remotely handled process equipment in the U.S. and Europe. ALSTEC and FWENC developed the NRC-approved ISFSI at Fort St. Vrain in Colorado. ALSTEC used a similar NRC-licensed design for the Paks Nuclear Power Station SNF dry storage facility in Hungary. ALSTEC has also worked closely with Foster Wheeler to design, deliver, and support start-up of the Multi-Canister Overpack Handling Machine at Hanford. UE, a sister company to Public Service Company of Colorado, is a full-service engineering firm dedicated to power plant design and operation. The UE group includes the core group of professionals who led the Fort St. Vrain decontamination and decommissioning efforts and who supervised the design, licensing, construction, and operation of the Fort St. Vrain ISFSI.

As stated earlier, FWENC has previously teamed with ALSTEC and UE to design, license, construct, and operate the Fort St. Vrain ISFSI in Colorado. In addition, FWENC has received NRC approval of a Topical Report for the generic Modular Vault Dry Storage system that was used at Fort St. Vrain. The Fort St. Vrain MVDS was designed by ALSTEC and FWENC, constructed by FWENC, and was originally operated by Public Service Company of Colorado.

FWENC will have and maintain an adequate complement of trained and certified personnel prior to receipt of SNF for storage and throughout the different phases of the project. The technical qualifications of the staff managing the design, construction, and operation of the ISF Facility are contained in Chapter 9 of the SAR, which also contains the organizational structure to be implemented.

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3.0 TECHNICAL INFORMATION - SAFETY ANALYSIS REPORT

FWENC was awarded a contract by the DOE to receive, transfer, package, and place into interim dry storage, SNF currently stored, or scheduled to be stored, at the INEEL located near Idaho Falls, Idaho. The specific fuel to be stored at the applicant facility consists of:

- Cores 1 and 2 from Peach Bottom Unit 1, a high temperature, gas cooled reactor that operated from March 1966 until October 1974.
- various reflector modules and rods from Shippingport, an experimental light water breeder reactor (LWBR) that ceased operation in 1983.
- SNF from various Training, Research, and Isotope reactors built by General Atomics (TRIGA).

TRIGA fuel assemblies account for less than 2 percent by weight of the total heavy metal content of SNF to be stored at the ISF Facility. A detailed description of this SNF is contained in Section 3.1.1 of the *Safety Analysis Report.*

The ISF Facility is a dry storage ISFSI, with facilities for the receipt and repackaging of the SNF into sealed storage canisters. The ISF Facility is a fully enclosed building complex that allows for year-round operations. The loaded and sealed storage canisters are placed in individual storage tubes that have both a bolted lid and double metallic O-ring seals, providing redundant confinement boundaries. The storage tubes are housed in concrete storage vaults that provide radiological shielding and passive natural convection air-cooling.

The *Safety Analysis Report* provides a summary description of the design and operations of the facility. This facility is designed for a service life of 40 years.

The SAR has been written using the guidance of NRC Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* dated August 1989 and meets the requirements contained in 10 CFR 72.24, *Contents of Application: Technical Information* (Refs. 5 and 1). FWENC also utilized the NRC's *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NUREG-1567) and associated Interim Staff Guidance documents in the preparation of the SAR (Ref. 6). FWENC is providing under separate cover a summary matrix illustrating how the SAR achieves compliance with the acceptance criteria provided in NUREG-1567 and associated Interim Staff Guidance.

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4.0 CONFORMITY TO GENERAL DESIGN CRITERIA

The ISF Facility is designed to receive, transfer, and provide dry storage and passive cooling for SNF currently in, or scheduled for, interim storage within the INEEL.

Title 10 CFR 72 Subpart F, *General Design Criteria* establishes the design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components identified as important to safety. These criteria are divided into the following areas: overall requirements, nuclear criticality safety, radiological protection, storage and handling, and decommissioning. The ISF Facility complies with these general design criteria as described in the SAR and other documents submitted with this application. The following table provides a cross-reference for each design criteria to the applicable SAR sections or other document provided with this application.

10 CFR 72 Section	Applicable SAR or Other Document Sections	
72.122(a) Quality Standards	3.4	Classification of Structures, Systems, and Components
	4.2.1	Structural Specifications – Storage Structures
	4.7.1	Structural Specifications – Spent Fuel Handling Operation Systems
	QPP	Quality Program Plan (ISF-FW-PLN-0017)
72.122(b) Protection against environmental conditions and natural phenomena	3.2	Structural and Mechanical Safety Criteria
	4.2.3.3	Design Bases and Safety Assurance – Storage Structures
	4.7.3.3	Design Bases and Safety Assurance – Spent Fuel Handling Operation Systems
	8.2.5	External Events
72.122(c) Protection against fires and explosions	3.3.6	Fire and Explosion Protection
	4.3.8	Fire Protection System
	8.2.4.4	Fire and Explosion
72.122(d) Sharing of structures, systems, and components	3.1.2.4	Utilities
	4.1.2.3	Site Utility Supplies and Systems
72.122(e) Proximity of sites	2.2	Nearby Industrial, Transportation, and Military Facilities
	8.2.5.6	Accidents At Nearby Sites
72.122(f) Testing and maintenance of systems and components.	3.4	Classification of Structures, Systems, and Components
	4.3.9	Maintenance Systems
	5.1.3.5	Maintenance Techniques
72.122(g) Emergency capability	3.3.8	Industrial and Chemical Safety
	4.3.8	Fire Protection System
	EP	Emergency Plan (ISF-FW-PLN-0021)

10 CFR 72 Section	Applicable SAR or Other Document Sections	
72.122(h) Confinement barriers and systems	3.3.2	Protection by Multiple Confinement Barriers and Systems
	4.2.2.3	Confinement Features - Storage Structures
	4.7.2.3	Confinement Features – Spent Fuel Handling Operation Systems
	6.2	Off-gas Treatment and Ventilation
72.122(i) Instrumentation and control systems	3.3.3.2	Instrumentation
	7.3.4	Area Radiation and Airborne Radioactivity Monitoring Instrumentation
	9.2	Preoperational Testing and Operation
72.122(j) Control room or control area	5.5	Control Room and Control Areas
72.122(k) Utility or other services	3.1.2.4	Utilities
	4.1.2.3	Site Utility Supplies and Systems
	5.3.1	Operating Systems
72.122(l) Retrievability	3.3.7.1	Spent Fuel or High-Level Radioactive Waste Handling and Storage
	4.2.3	Individual Fuel Storage System Unit Description
	4.7.3.3	Design Bases and Safety Assurance – Spent Fuel Handling Operations Systems
72.124(a) Design for criticality safety	3.3.4	Nuclear Criticality Safety
	4.2.3.3.7	Criticality Evaluation – Storage Structures
	4.7.3.4	Criticality Evaluation for Spent Fuel Handling Operations
72.124(b) Methods of criticality control	3.3.4.1	Control Methods for Prevention of Criticality
	4.2.3.3.7	Criticality Evaluation – Storage Structures
	4.7.3.4	Criticality Evaluation for Spent Fuel Handling Operations
	5.1.3.1	Criticality Prevention
72.124(c) Criticality Monitoring	3.3.5.3	Radiological Alarm Systems
	7.3.4	Area Radiation and Airborne Radioactivity Monitoring Instrumentation
72.126(a) Exposure Control	3.3.5	Radiological Protection
	5.1	Operation Description (primarily contamination control for various operations)
	7.3	Radiation Protection Design Features
72.126(b) Radiological alarm systems	3.3.5.3	Radiological Alarm Systems
	7.3.4	Area Radiation and Airborne Radioactivity Monitoring Instrumentation
72.126(c) Effluent and direct radiation monitoring	3.3.5.3	Radiological Alarm Systems
	7.6.1	Effluent and Environmental Monitoring Program

10 CFR 72 Section	Applicable SAR or Other Document Sections
72.126(d) Effluent control	3.3.7.2 Radioactive Waste Treatment 6.0 Generated Waste Confinement and Management
72.128(a) Spent fuel and high-level radioactive waste storage and handling systems	3.3.7.1 Spent Fuel or High-Level Radioactive Waste Handling and Storage 4.2.3.3 Design Bases and Safety Assurance – Storage Structures 4.7.3.3 Design Bases and Safety Assurance – Spent Fuel Handling Operation Systems
72.128(b) Waste treatment	3.3.7.2 Radioactive Waste Treatment 6.0 Generated Waste Confinement and Management
72.130 Criteria for Decommissioning	Proposed Decommissioning Plan (ISF-FW-PLN-0027) 3.5 Decommissioning Considerations, 9.6 Decommissioning Plan

Appendix A to this License Application contains a request for an exemption from certain specific regulatory requirements related to the design requirements for withstanding the effects of earthquakes as required by 10 CFR 72.122(b), "*Protection against environmental conditions and natural phenomena*." This exemption request is filed in accordance with 10 CFR 72.7, "*Specific exemptions*", and seeks exemption from requirements contained in 10 CFR 72.102(f)(1) related to the specified design earthquake for the ISF Facility. As discussed in the exemption request, FWENC proposes to use a probabilistic seismic hazard analysis in lieu of the deterministic methods specified by 10 CFR 72.102(f)(1). The proposed methodology is consistent with current NRC requirements in Parts 50 and 100 for new nuclear power plants and in Part 60 for geological radioactive waste repositories and current proposed revisions to Part 72.

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5.0 OPERATING PROCEDURES; ADMINISTRATIVE AND MANAGEMENT CONTROLS

Written procedures will be developed, implemented, maintained, and utilized for ISF Facility operations, maintenance, and testing activities identified as important to safety. Chapter 5 of the SAR provides operational guidance for those tasks necessary to ensure safe and reliable operation during normal and potential off-normal events during both initial fuel repackaging and subsequent storage operations.

Chapter 9 of the SAR describes FWENC's organizational structure and approach to implementing the basic managerial and administrative controls described in ANSI N299-1976.

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6.0 QUALITY ASSURANCE PROGRAM

Activities associated with design, fabrication, construction, testing, operation, modifications, and decommissioning of the structures, systems, and components of the ISF Facility are conducted in accordance with a quality assurance program as described in the *Quality Program Plan*, ISF-FW-PLN-0017. FWENC will ensure that the provisions of the Quality Program Plan and its implementation are understood by the personnel involved in its execution. This program is applied to the design, procurement, construction, testing, operation, modification, and decommissioning, if applicable, of ISF Facility structures, systems, and components important to safety. Adherence to this program assures compliance with the requirements of 10 CFR 72, Subpart G.

FWENC previously submitted the QPP to the NRC via letter dated March 31, 2001(Ref. 3). The FWENC Nuclear Quality Assurance Program is described in Chapter 11 of the SAR.

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7.0 OPERATOR TRAINING

Operation of equipment and controls that have been identified as important to safety in the SAR will be limited to trained and certified personnel or will be conducted under the direct visual supervision of trained and certified personnel. The *Operator Training and Certification Plan* (ISF-FW-PLN-0031) describes the training program that will be used to certify these personnel.

Details of the training program for personnel performing ISF Facility security functions are provided under separate cover in the *Physical Protection Plan* (ISF-FW-PLN-0029).

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8.0 INVENTORY AND RECORDS REQUIREMENTS

Material balances and inventories of the Peach Bottom Cores 1 and 2, Shippingport light water breeder water (LWBR) fuel, and TRIGA fuel from various sources to be stored in the ISF Facility will be performed. Records of the SNF will be maintained in accordance with 10 CFR 72.72, *Material Balance, Inventory, and Records Requirements for Stored Materials*. Section 9.4.2 of the SAR describes the records management system, including the provision to maintain records on the identity of the SNF stored at the ISF Facility. A description of the material inventory and records system to be used at the ISF Facility is provided below.

Records in the form of paper copies will be kept to show the receipt, inventory, location, and transfer of all Peach Bottom Cores 1 and 2, Shippingport LWBR fuel, and TRIGA fuel from various sources, at or from the ISF Facility. The records will include the estimated quantity of material contents of each canister, including the estimated special nuclear material in each canister (based on original loading quality assurance documentation), ISF canister identification, and storage location within the ISF Facility. These records will also include the movements of each canister to or within the ISF Facility, and movements away from the ISF Facility to a permanent or other interim storage facility. These records will be kept for as long as the SNF is stored at the ISF Facility and will be transferred along with the canisters. A duplicate set of these records will be kept at a separate location away from the location of the principal records, sufficiently remote that a single event will not destroy both sets of records. This duplicate set of records will be kept for a period of 5 years after the canisters have been removed from the ISF Facility.

Administrative controls and labeling of the ISF canisters will be utilized to maintain accurate records of material location. Each ISF canister will be labeled with a unique identifier. Information, including location, on all ISF canisters will be documented and kept with other ISF Facility records. Prior to any movement of a canister, within or from the ISF Facility, established procedures will require a review of the documentation to help ensure that the proper canister is being moved.

In addition, a physical inventory of the canisters at the ISF Facility will be performed annually. Records will be kept of the results of the current inventory and retained until termination of the NRC license. Physical inventories will be performed in accordance with written procedures and will consist primarily of confirmation that all the canisters are in their assigned locations by showing there is no evidence of tampering with the seals of the storage tubes. In addition to the inventory procedures, other written material control and accounting procedures will be prepared and implemented, as necessary, to account for radioactive material in storage. Copies of current material control and accounting procedures will be retained until termination of the NRC license.

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9.0 PHYSICAL PROTECTION

The physical protection program for the ISF Facility has been developed in accordance with 10 CFR 72, Subpart H. This program is described in ISF Facility *Physical Protection Plan*, ISF-FW-PLN-0029. The *Physical Protection Plan* also incorporates the guard training and qualification plan and the safeguards contingency plan required by 10 CFR 72.180 and 72.184. The program contains safeguards information and is protected and controlled in accordance with and 10 CFR 73.21 (Ref. 7). The *Physical Protection Plan* is being submitted under a separate cover.

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10.0 DECOMMISSIONING PLAN

In accordance with the contract between DOE and FWENC for the construction and operation of the ISF Facility, the DOE retains responsibility for the ultimate decommissioning of the facility (Ref. 4). To support planning for this effort, FWENC has prepared a cost estimate for decommissioning and included this estimate in the *Proposed Decommissioning Plan*.

The *Proposed Decommissioning Plan*, ISF-FW-PLN-0027, provides the information required by 10 CFR 72.30, *Financial Assurance and Recordkeeping for Decommissioning*. The proposed plan for decommissioning addresses the decontamination and dismantling of equipment and structures, and site restoration. The first phase, decontamination and dismantling, will begin after all SNF has been transferred from the ISF Facility. The Site Restoration Phase will begin immediately after the decontamination and dismantling is complete. A final decommissioning plan will be submitted prior to the start of decommissioning work in accordance with 10 CFR 72.54(d).

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11.0 EMERGENCY PLAN

FWENC has developed an emergency plan for the ISF Facility that interfaces with the current *INEEL Emergency Plan/RCRA Contingency Plan* (referred to as the INEEL Base Plan) (Ref. 8). The *Emergency Plan,* ISF-FW-PLN-0021, provides the information required by 10 CFR 72.32, *Emergency Plan.* Additionally, FWENC has prepared an appendix to the *Emergency Plan* to indicate where the requirements of 10 CFR 72.32(b) and Interim Staff Guidance Document 16 are implemented in the INEEL Base Plan and the *Emergency Plan.*

Primary ISF Facility emergency response will be provided by DOE and its management and operating contractor personnel located at the INEEL. In accordance with 10 CFR 72.32 (b) (14), the ISF Facility *Draft Emergency Plan* was provided to DOE-ID and its M&O contractor for sixty (60) day review (Ref. 9). The ISF Facility *Emergency Plan,* submitted as a part of this License Application, has been modified accordingly to reflect resolution of these comments. Copies of the comments received from offsite response organizations are included with this document as required by 10 CFR 72.32(b)(14).

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12.0 ENVIRONMENTAL REPORT

The *Environmental Report* for the ISF Facility is included as required by 10 CFR 72.34. This report describes the potential environmental impacts associated with the construction and operation of the ISF Facility. The *Environmental Report* concludes that the construction and operation of the ISF Facility will not result in significant adverse environmental effects. This conclusion is consistent with the NRC's generic finding in 10 CFR 51.23 that spent fuel may be stored at such an ISFSI for a period of at least 30 years without significant environmental impact (Ref. 10). In addition, the minor environmental effects associated with the construction and operation of the ISF Facility are well within those previously evaluated in the *DOE Programmatic Spent Fuel Management and Idaho National Engineering Laboratory Restoration and Waste Management Programs Final Environmental Impact Statement* (Ref. 11).

The *Environmental Report* meets the requirements set forth in 10 CFR 72.34, *Environmental Report*; 10 CFR Part 72, Subpart E, *Siting Evaluation Factors*; and 10 CFR 51, Subpart A, *National Environmental Policy Act—Regulations Implementing Section 102(2)*.

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13.0 PROPOSED LICENSE CONDITIONS

The proposed license conditions, entitled *Proposed Technical Specifications,* are included as an appendix to this License Application. FWENC developed these *Proposed Technical Specifications* consistent with the guidance contained in NUREG-1745, *Standard Format and Content for Technical Specifications for 10 CFR 72 Cask Certificates of Compliance* (Ref. 12) with modifications as appropriate to accommodate a site-specific license and unique features of the ISF Facility. These *Proposed Technical Specifications* include restrictions that limit facility conditions consistent with the ISF Facility design requirements for the safe receipt, handling, and interim storage of SNF.

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14.0 REFERENCES

1. Title 10, Code of Federal Regulations, Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.*

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.50, Revision 1, *Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste,* September 1989.

3. Letter dated March 31, 2001, from Mr. Robert K. Stampley to the U.S. Nuclear Regulatory Commission (FW-NRC-ISF-01-0144) transmitting FWENC Quality Program Plan, ISF-FW-PLN-0017.

4. DOE-ID (2000), *Contract Award and Notice to Proceed, Contract No. DE-AC07-001D13729, Spent Nuclear Fuel Dry Storage Project,* U.S. Department of Energy, Idaho Falls, Idaho, May.

5. U.S. Nuclear Regulatory Commission. Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Installation or Monitored Retrievable Storage Installation (Dry Storage),* August 1989.

6. U.S. Nuclear Regulatory Commission, NUREG 1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities,* March 2000.

7. Title 10, Code of Federal Regulations, Part 73, *Physical Protection of Plants and Materials.*

8. DOE-ID *INEEL Emergency Plan/Resource Conservation and Recovery Act (RCRA) Contingency Plan.*

9. Letter dated September 20, 2001, from Mr. Ron Izatt (FW-DOE-ISF-0541) to L. Johnson, DOE, transmitting FWENC Emergency Plan, ISF-FW-PLN-0021.

10. Title 10, Code of Federal Regulations, Part 51, *Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.*

11. DOE (1995), *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement,* DOE/EIS-0203-F, April.

12. U.S. Nuclear Regulatory Commission, NUREG 1745, *Standard Format and Content for Technical Specifications for 10 CFR 72 Cask Certificates of Compliance,* June 2001.

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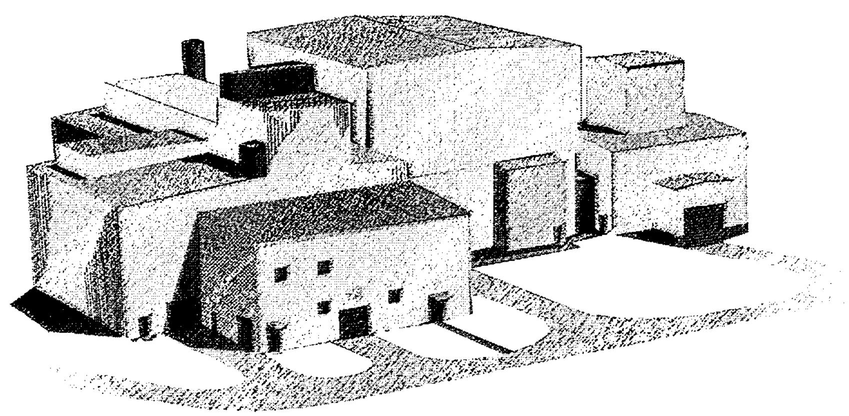
License Application

Appendix A

Request for Exemption from Seismic Design Requirement

Idaho Spent Fuel Facility

Docket No. 72-25



FOSTER WHEELER ENVIRONMENTAL CORPORATION

Appendix A
Request for Exemption from Seismic Design Requirement

1.0 BACKGROUND

The purpose of this Appendix is to request an exemption pursuant to 10 CFR 72.7, to 10 CFR 72.102(f)(1) (Ref. 1) for the Idaho Spent Fuel (ISF) Facility located at the Idaho National Engineering and Environmental Laboratory (INEEL) adjacent to the Idaho Nuclear Technology and Engineering Center (INTEC) site. If approved, the exemption would change the methodology for calculating the design earthquake (DE), used for the design of structures at the ISF Facility, from a deterministic approach to a probabilistic, risk-informed approach. The exemption would not endanger life or property or the common defense and security and would be in the public interest.

10 CFR 72.102(b) requires Independent Spent Fuel Storage Installation (ISFSI) sites west of the Rocky Mountain Front to evaluate seismicity by the techniques of 10 CFR 100 - Appendix A (Ref. 2). For sites that have been evaluated under the criteria of Appendix A, 10 CFR 72.102(f)(1) requires the DE to be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant. 10 CFR 100 - Appendix A uses a deterministic approach for determining the SSE (i.e., the DE).

Foster Wheeler's request for an exemption and proposed methodology for evaluating seismic hazards is consistent with recent revisions to 10 CFR Part 50 and Part 100 requirements for power reactors; 10 CFR Part 60 (Ref. 3) requirements for high-level radioactive waste geologic repositories; and NRC plans to amend 10 CFR Part 72 to allow the use of probabilistic seismic hazards analysis (PSHA) for an independent spent fuel storage installation (ISFSI). This request is also consistent with previous exemptions granted by the NRC for similar facilities:

- This exemption request is similar to the U.S. Department of Energy - Idaho Operations Office (DOE-ID) exemption granted regarding the seismic design requirements of 10 CFR 72.102(f)(1) for the TMI-2 ISFSI. The TMI-2 ISFSI is also located at the INEEL in the INTEC site. SECY-98-071 (Ref. 4) and the Federal Register (64 FR13828, 13829) (Ref. 5) documents NRC approval of the DOE-ID exemption request.
- In addition, this exemption request is similar to the Private Fuel Storage Limited Liability Corporation (PFSLLC) exemption request from the same seismic design requirements of 10 CFR 72.102(f)(1) for the PFS Facility that will be located at Skull Valley, Utah. The SER for the PFS Facility documents the NRC staff's approval of the PFSLLC exemption request.

FWENC has determined that there is an adequate safety basis for an exemption to the requirements of 10 CFR 72.102(f)(1), supported by a site-specific radiological risk analysis. The approval of this exemption request would be consistent with NRC policy and regulations applicable to other facilities (i.e., nuclear power plants and high-level waste geologic repositories) that carry greater risk than a facility licensed to 10 CFR Part 72. Considering the minor radiological consequences of seismic activity induced accidents analyzed at the ISF Facility, FWENC considers the proposed method for calculating the DE to be acceptable.

2.0 REGULATORY BASIS

Relevant and Related Regulations

Regulation 10 CFR 72.102(b) requires ISFSI sites west of the Rocky Mountain Front, such as the ISF Facility site, to have seismicity evaluated by the techniques of 10 CFR 100 – Appendix A, also known as a deterministic seismic hazard analysis (DSHA). In addition, 10 CFR 72.102(f)(1) states that: "For sites that have been evaluated under the criteria of appendix A of 10 CFR part 100, the DE must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant." In this context, the DE refers to the design peak ground acceleration (PGA), with an appropriate response spectrum, caused by the largest credible earthquake that could affect a site without regard for the probability of this event through time.

On January 10, 1997, 10 CFR Parts 50 (Ref. 6) and 100 were revised to allow the use of the PSHA methodology to address uncertainties inherent in determining nuclear power plant seismic design values. These revisions were accomplished through the addition of 10 CFR 50 – Appendix S and 10 CFR 100.23. The PSHA method considers the frequency, as well as magnitude, of earthquakes that may affect a site. Rather than base the seismic design on the largest ground motion likely to ever affect a site, a PSHA derives a site-specific hazard curve showing ground motion level versus annual probability of exceedence or, inversely, ground motion return period. The NRC issued Regulatory Guide 1.165 (Ref. 7) to provide guidance on calculation of the DE using PSHA techniques. However, because 10 CFR 72.102 currently requires seismicity to be evaluated by the deterministic technique of 10 CFR 100 – Appendix A, absent an exemption, applicants for ISFSI licenses are not able to utilize the PSHA methodology that 10 CFR 100.23 and 10 CFR 50 - Appendix S promulgates.

On January 3, 1997, 10 CFR Part 60 was revised to permit the use of probabilistic, risk-informed methodology in designing for hazards (including seismic) at high-level waste geologic repositories.

On June 4, 1998, the NRC staff issued SECY-98-126 (Ref. 8). In SECY-98-126, the NRC staff proposed to modify the 10 CFR Part 72 seismic requirements to a level commensurate with the risk of an ISFSI by providing for the use of PSHA methodology. In accordance with the Commission's risk-informed approach to regulatory decision making, the NRC staff recommended proceeding with rulemaking per "Option 3". Under this option, new Part 72 licensees would be required to conform to 10 CFR 100.23 and allow a graded approach to seismic design for ISFSI structures, systems, and components (SSC). This graded approach would be in lieu of Section 72.102(f)(1).

On September 26, 2001, the NRC staff issued SECY-01-0178 (Ref. 9). In SECY-01-0178, the NRC staff proposed to add another option to the recommendations made in SECY-98-0126. The new option provides an alternative to the graded approach for the seismic design of dry cask ISFSI SSCs. The option also recommends the creation of a new section in Part 72 that is based on 10 CFR 100.23, instead of referencing 10 CFR 100.23. Additionally, the new option reversed the previous graded approach option with the maintenance of the current Part 72 approach of a single design basis event because of the relative simplicity of the ISFSI design and operation.

The NRC staff logic for acceptance of the additional option was based on a comparison to nuclear power plants. An operating ISFSI facility is a relatively simple facility in which the primary activities are spent fuel receipt, handling, and storage. An ISFSI facility does not have the variety and complexity of active systems necessary to support an operating nuclear power plant. After the spent fuel is in place, an ISFSI facility is a static operation. During normal operations, the conditions required for the release and dispersal of significant quantities of radioactive materials are not present. The likelihood of release and dispersal of radioactive materials is low primarily due to low heat generation rates of spent fuel with greater than the required one year of decay before storage in an ISFSI, combined with low inventory of volatile radioactive materials readily available for release to the environs. The radiological risk associated with an ISFSI facility is significantly less than the risk associated with a nuclear power plant, and therefore, the use of a lower design earthquake ground motion is appropriate.

Relevant Previous Exemption Requests

This exemption request is similar to the U.S. Department of Energy - Idaho Operations Office (DOE-ID) exemption granted regarding the seismic design requirements of 10 CFR 72.102(f)(1) for the TMI-2 ISFSI. The TMI-2 ISFSI is also located at the INEEL in the INTEC site. SECY-98-071 and the Federal Register (64 FR13828, 13829) documents NRC approval of the DOE-ID exemption request.

On September 15, 1998, the Idaho Operations Office of the Department of Energy (DOE-ID) submitted a 10 CFR 72.7 exemption to the seismic requirements of 10 CFR 72.102(f)(1) for the TMI-2 ISFSI located on the INEEL in Idaho. As documented in the NRC staff's evaluations of the requested exemptions for the TMI-2 ISFSI (Ref. 4 and Ref. 5) and the PFS ISFSI (Ref. 10):

- In the Statement of Consideration accompanying the initial 10 CFR Part 72, the NRC recognized the reduced radiological hazard associated with dry cask ISFSI and stated that the seismic design basis ground motions for these facilities need not be as high as for commercial nuclear power plants.
- The NRC staff accepted the probabilistic, risk-graded approach to seismic hazard characterization and design that is described in the DOE Standard 1020-94 (Ref. 12). The DOE Standard 1020-94 requires facilities, which have a potential accident consequence similar to an ISFSI, to be designed for ground motion that has a mean recurrence interval of 2000 years.
- The NRC staff considered the seismic design philosophy in 10 CFR Part 60 (as revised on January 3, 1997) for high-level waste repository surface facilities. The Part 60 revision established an NRC precedent by accepting a risk-graded approach in licensing a facility that is similar to an ISFSI in terms of radioactive material present and potential accident scenarios. The NRC staff noted that the seismic design philosophy in Part 60 is based on a PSHA and is similar in approach to that presented in DOE Standard 1020-94 (Ref. 12).

For the TMI-2 ISFSI, the NRC staff concluded (Refs. 4 and 5) that a DE value that envelopes the 2000-year return period value using PSHA methodology is acceptable. This NRC staff determination was primarily based upon the following information:

- The radiological consequences from beyond-design basis accidents leading to cask or canister rupture (i.e. non-mechanistic confinement boundary failure) is well below the 5 rem total effective dose equivalent (TEDE) limit of 10 CFR 72.106(b).
- The risk-graded approach to seismic hazard characterization and design in DOE Standard 1020-94 (Ref. 12), which is similar to the risk-graded approach of using the 2000-year return period mean ground motion as the DE, is adequately conservative.

- The expected life span of the ISFSI, 20 years with the possibility of renewal (per 10 CFR 72.42), justifies the 2000-year return period mean ground motion as the DE.

In addition, this exemption request is similar to the Private Fuel Storage Limited Liability Corporation (PFSLLC) exemption request from the same seismic design requirements of 10 CFR 72.102(f)(1) for the PFS Facility that will be located at Skull Valley, Utah. On April 2, 1999, PFSLLC submitted a 10 CFR 72.7 exemption to the seismic requirements of 10 CFR 72.102(f)(1). The SER for the PFS Facility documents the NRC staff's approval of the PFSLLC exemption request.

For the PFS ISFSI, the NRC staff concluded (Ref. 10) that a DE value that envelopes the 2000-year return period value using PSHA methodology is acceptable. This NRC staff determination was based (in part) upon the following information:

- The PFS ISFSI design is functionally similar to the TMI-2 ISFSI.
- The NRC has accepted a DE value that envelops the 2000-year period probabilistic ground motion value for the TMI-2 ISFSI license.

The ISF facility is also a dry spent fuel storage facility licensed to 10 CFR 72. Therefore, the FWENC proposed exemption to 10 CFR 72.102(f)(1) applies the same approach for establishing the DE at the ISF Facility as that accepted by the NRC for the TMI-2 and PFS ISFSIs. Foster Wheeler Environmental Corporation (FWENC) understands that the NRC evaluated these exemptions against DOE Standard 1020-94 (Ref. 12), *"Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities,"* and the more recent NRC positions previously noted. The DOE standard takes a graded approach to designing critical facilities, requiring facilities with greater accident consequences to use higher design requirements for phenomena such as earthquakes. For the TMI-2 and PFS facilities, the NRC staff accepted a design earthquake value that envelopes the 2,000-year return period value using PSHA methodology.

The use of probabilistic techniques and a risk-informed approach are also compatible with the direction provided by the NRC in Direction Setting Issue 12, *"Risk-Informed, Performance-Based Regulation."*

While the NRC has indicated in SECY-98-126 that it plans to amend 10 CFR 72.102 to permit use of PSHA methodology and a risk-informed approach to calculate the DE at ISFSI sites, this rulemaking may not be completed before issuance of the ISF Facility license. Therefore, FWENC is requesting an exemption to 10 CFR 72.102(f)(1). The exemption would permit the DE for the ISF Facility to be calculated using the more recent PSHA methodology, in accordance with the guidance in Regulatory Guide 1.165, and allow the risk-informed approach of 10 CFR Part 60, which the NRC staff approved for the TMI-2 ISFSI at the INTEC (Idaho) and the PFS Facility at Skull Valley (Utah).

3.0 EVALUATION

Radiological Hazards Associated with the ISF Facility

A FWENC evaluation analyzed hypothetical, beyond-design basis, non-mechanistic failures of the ISF canister confinement boundary and the ISF transfer area confinement boundary (SAR Section 8.2.4.5). The worst-case result from each of these hypothetical beyond-design basis events was used to bound the off-site consequences from seismic induced failures during each phase of the ISF Facility repackaging operation and subsequent interim storage of SNF.

The ISF Facility repackages fuel from DOE containers into an ISF canister for interim storage. The ISF canister is seal welded and leak tested to ensure integrity and represents the first configuration that was analyzed for hypothetical off-site doses from a non-mechanistic failure. This evaluation assumes the canister is internally contaminated from failed fuel and the contamination is released to the environment continuously for 30 days. This scenario is conservative as there is no realistic mechanism for the particulate contamination to be released from the ISF canister. Particulate contamination from spent fuel is the largest contributor to dose consequences from this hypothetical accident.

The hypothetical accident conditions evaluated involved failure of 100% of the coating/cladding on each fuel assembly. The ISG-5 (Ref. 11) release fractions were applied to calculate the source term available for release. The appropriate ANSI/ANS-5.10-1988 (Ref. 13) airborne release fraction and the maximum allowed canister leak rate were applied to determine the activity release rate for the event. This analysis conservatively assumed that the receptor individual was continuously located at the INEEL controlled area boundary for the duration of the event.

The FWENC evaluation concluded that the total effective dose equivalent (TEDE) from this beyond-design basis accident to the off-site maximally exposed individual (MEI) was calculated to be 0.003 mrem. Of the SNF types handled at the ISF Facility, an ISF canister with ten Peach Bottom 2 (PB-2) fuel assemblies was identified as bounding for off-site dose consequences. This represents the worst-case interim storage phase canister handling operation because the sealed ISF canister will only temporarily be located outside of a sealed storage tube that provides further protection.

When the ISF Facility is processing SNF in the Fuel Packaging Area (FPA), each fuel assembly is handled outside of a container or canister. The confinement barrier during this phase of operation is provided by the building integrity including HEPA filters inside the FPA on both the intake and exhaust ducting. This second beyond-design basis evaluation was performed by postulating a non-mechanistic breach of one HEPA filter after it becomes coated with radioactive contaminants up to the administrative control limit of 250 mR/hour (approximately 100 mCi) on face of the filter. This evaluation assumes a non-mechanistic event releases the contamination to the environment. This analysis conservatively assumed that the release occurred over a 2 hour period and the receptor individual was continuously located at the INEEL controlled area boundary for the duration of the event.

The hypothetical accident conditions for each fuel type were evaluated. The activity on the filter was from a single fuel type and the applicable ANSI/ANS-5.10-1988 release fraction was used to determine the total activity released to the environment. The total activity released from the filter was conservatively assumed to be respirable.

The FWENC evaluation concluded that the TEDE from this beyond-design basis accident to the off-site MEI was calculated to be 0.02 mrem. Of the SNF types handled at the ISF Facility, Peach Bottom 2 fuel was identified as bounding. This represents the worst-case ISF repackaging phase of the ISF Facility operation and is the bounding case for ISF operations. This dose is well below the 5,000 mrem TEDE siting evaluation factor of 10 CFR 72.106(b).

The dose consequences from the hypothetical non-mechanistic accident cases evaluated for the ISF Facility during storage operations (0.003 mrem TEDE to the MEI) and fuel packaging operations (0.02 mrem TEDE to the MEI) are lower than those estimated for the TMI-2 ISFSI (75 mrem) and the PFS ISFSI (74.9 mrem). The NRC staff have previously accepted a PSHA based on a seismic event with a 2,000-year return period for these ISFSIs. The results of Foster Wheeler's analysis for the ISF Facility indicate that the dose consequences from bounding accident scenarios are no greater than those estimated for the TMI-2 and PFS ISFSIs, and are significantly below the regulatory limits established by 10 CFR Part 72. Therefore, Foster Wheeler concludes that a PSHA based on a seismic event with a 2,000 year return period would be acceptable for the ISF Facility.

ISF Facility PSHA Methodology

Foster Wheeler's methodology for developing the PSHA for the ISF Facility is based on existing seismic data for the INEEL site, as summarized below and in Chapter 2 of the ISF Facility Safety Analysis Report (SAR). This methodology is described below, and the resulting seismic motions compared to both deterministic and probabilistic seismic motions calculated for the nearby TMI-2 facility, and for the INEEL site.

A deterministic seismic hazards analysis (DSHA) has not been performed for the ISF Facility. However, a DSHA has been performed for the TMI-2 ISFSI site, located approximately 1,200 feet west of the ISF site and those results are presented here for comparison. Woodward-Clyde Federal Services performed the DSHA for the TMI-2 ISFSI site in accordance with the requirements of 10 CFR 72.102(f)(1). The results of this analysis are documented in the Safety Analysis Report for the TMI-2 ISFSI (Ref. 14). This evaluation yielded a peak horizontal acceleration of 0.28 g at bedrock. The deterministically derived design earthquake would be defined by an appropriate response spectrum scaled to 0.28 g for the rock.

The NRC staff granted an exemption to the DOE-ID for the TMI-2 ISFSI and found that a design earthquake defined by a 0.30 g peak ground acceleration (PGA) with a 2,000-year return period generated by the probabilistic method to be acceptable. The 1996 probabilistic study that calculated the 0.30 g PGA also defined the peak horizontal acceleration at bedrock to be 0.13 g for a 2,000-year return period event.

DOE-ID recently issued INEEL/EXT-99-00775, "Development of Probabilistic Design Basis Earthquake (DBE) Parameters for Moderate and High Hazard Facilities at INEEL," (Ref. 15). This report was based on a detailed site-specific PSHA performed by URS Greiner Woodward-Clyde Federal Services in 1999 and 2000 (Refs. 16 and 17) which followed the methodology of Regulatory Guide 1.165 and DOE Standard 1020. The results of the PSHA yielded peak horizontal accelerations at INTEC of 0.123 g at bedrock for a design earthquake with a 2,500-year return period.

Ground motions for a 2,500-year return period event were calculated instead of 2,000-year return to account for anticipated changes in DOE Standards that will require a 2,500-year return period for Performance Category 3 type facilities. FWENC has chosen to use a 2,500-year return period as the basis for the PSHA of the ISF Facility, versus 2,000 years used for the TMI-2 and PFS ISFSIs, to incorporate the available data from this latest evaluation.

For the ISF Facility, the 0.123 g bedrock ground motion was used as the controlling design motion. This motion was propagated to the soil surface using site-specific soil properties and the methodology described in Chapter 2 of the ISF Facility Safety Analysis Report. A mean response was calculated as part of the site response analysis.

Using the probabilistic methodology and the rock and soil data described above, the DE at rock for the ISF Facility is 0.123 g in the horizontal direction and the equivalent soil surface PGA is 0.19 g. The DSHA earthquake calculated by Woodward-Clyde for the TMI-2 ISFSI was 0.28 g at bedrock and 0.56 g at the soil surface. These values would be applicable to the ISF site if a deterministic approach was taken.

4.0 EVALUATION CONCLUSIONS

FWENC has identified a DSHA (per Appendix A of 10 CFR Part 100) from the TMI-2 ISFSI that is applicable to the ISF Facility and developed a PSHA (per 10 CFR 100.23) for the ISF Facility site. The FWENC assessments have determined that 0.28 g peak horizontal acceleration at bedrock by the deterministic method and 0.123 g peak horizontal acceleration DE at bedrock with a 2,500-year return period by the probabilistic method are appropriate.

The FWENC proposed DE for the ISF facility exceeds the PSHA methodology value for the 2,000-year return period mean ground motion and will maintain adequate design margin for seismic events. This margin provides additional assurance that public health and safety will not be adversely impacted.

The ISF facility DE has been determined in accordance with the latest probabilistic methodology using the risk-informed approach determined to be acceptable in:

- 10 CFR Part 60 for the pre-closure facilities of a geological repository for high-level radioactive waste,
- The NRC staff's preferred option for amending the seismic design requirements of 10 CFR Part 72, as described in SECY-98-126 (Ref. 8) and SECY-01-178 (Ref. 9).
- The NRC staff evaluation (Ref. 4 and Ref. 5) of the TMI 2 ISFSI, and
- The NRC staff evaluation (Ref. 10) of the PFS Facility.

Provided herein is sufficient information to demonstrate that its approval is reasonable, and more appropriate for a facility of this type. To require the analysis method presently required 10 CFR 72.102(f)(1) for establishing a 10 CFR Part 72 facility's DE would be an unnecessary burden. In addition, the FWENC proposed DE is consistent with previous NRC actions, and does not adversely effect public health and safety. Therefore, the NRC can reasonably conclude that an exemption from the requirements of 10 CFR 72.102(f)(1) the ISF Facility pursuant to 10 CFR 72.7 is appropriate.

5.0 REFERENCES

1. Title 10, Code of Federal Regulations, Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.*

2. Title 10, Code of Federal Regulations, Part 100, Reactor Site Criteria.

3. Title 10, Code of Federal Regulations, Part 60,Disposal of High-Level Radioactive Wastes in Geologic Repositories.

4. SECY-98-071, "Exemption to 10 CFR 72.102(f)(1) Seismic Design Requirement for Three Mile Island Unit 2 Independent Spent Fuel Storage Installation," April 8, 1998

5. Federal Register March 22, 1999 (Volume 64, Number 54, Page 13828-13829).

6. Title 10, Code of Federal Regulations, Part 50, Domestic Licensing of Production and Utilization Facilities.

7. NRC Regulatory Guide 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Motion," March 1997

8. SECY-98-126, "Rulemaking Plan: Geological and Seismological Characteristics for Siting and Design of Dry Cask Independent Spent Fuel Storage Installations, 10 CFR Part 72," June 4, 1998

9. SECY-01-178, "Modified Rulemaking Plan: 10 CFR 72 – Geological and Seismological Characteristics for Siting and Design of Dry Cask Independent Spent Fuel Storage Installations," September 26, 2001.

10. NRC Safety Evaluation Report (SER) Concerning the Private Fuel Storage Facility, Chapter 2, "Site Characteristics, October 18, 2000

11. Interim Staff Guidance (ISG) -5, Revision 1, "Confinement Evaluation"

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13. ANSI N510, Testing of Nuclear Air Treatment Systems. 1989.

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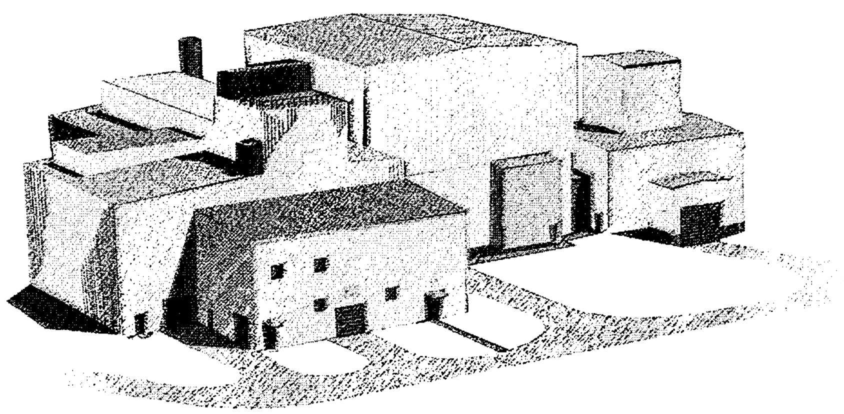
License Application

Appendix B

Operator Training and Certification Plan

Idaho Spent Fuel Facility

Docket No. 72-25



ISF-FW-PLN-0031

FOSTER WHEELER ENVIRONMENTAL CORPORATION

Contents

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1.0 INTRODUCTION

Subpart I, *Training and Certification of Personnel*, of Title 10 Code of Federal Regulations (CFR) 72, *Licensing Requirements for the Independent Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)*, (Ref. 1) requires that operation of equipment and controls identified as important to safety be limited to trained and certified personnel or be conducted under the direct visual supervision of an individual trained and certified.

The Operator Training and Certification Plan contained herein describes the initial certification training and re-certification plans to be implemented by Foster Wheeler Environmental Corporation (FWENC) to assure operation of the ISF Facility is performed in a manner consistent with protecting the health and safety of the public.

This training plan describes the personnel to whom the plan applies, the areas in which training is provided, what constitutes certification, how certification is maintained, and required qualifications (e.g. medical).

The recommendations provided in Draft Regulatory Guide Task HF 608-4 dated March 1982 (Ref. 2) were reviewed and used as input to the Operator Training and Certification Plan. Training will be developed, implemented, and maintained using the systematic approach to training (SAT).

This training plan becomes effective upon approval of the Operator Training and Certification Plan by the Nuclear Regulatory Commission (NRC).

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2.0 PLAN OVERVIEW

The Operator Training and Certification Plan includes an initial training course and biennial re-certification. Successful completion of this training plan is required for personnel that supervise and/or operate equipment and controls that have been identified as important to safety in the *Safety Analysis Report* (SAR) (Ref. 3).

As required by 10 CFR 72.194, the physical condition and general health of personnel certified for operation of equipment and controls that are important to safety cannot cause operation errors that could endanger other in-plant personnel or the public health and safety. This requirement will be satisfied by an initial and biennial physical examination.

The ISF Facility Manager is the certifying authority and shall designate in writing those personnel that are authorized to supervise and/or operate equipment and controls that have been identified as important to safety in the SAR.

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3.0 DEFINITIONS

Certified Operator – an individual certified to direct and/or operate equipment and controls important to safety in accordance with the requirements of Section 6.1.4, *Qualification* and maintains currency in accordance with Section 6.2.5, *Maintenance of Certified Operator Qualifications*.

On-the-Job Training (OJT) – training that is job performance oriented, leads to task qualification, and is conducted in a work environment.

Subject Matter Expert (SME) – an individual with the requisite skills or knowledge associated with a job/task or subject that has been designated by the ISF Facility Manager or designee to develop or perform training in specific areas.

Systematic Approach to Training (SAT) –performance-based process for development and evaluation of training requirements as provided in ANSI/ANS-3.1 1993, Section 6.2.1 (Ref. 4).

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4.0 RESPONSIBILITIES

4.1 ISF FACILITY MANAGER

The ISF Facility Manager has overall responsibility for the certification of persons that supervise and/or operate equipment and controls important to safety. The Administrative Services Manager and Operations Manager report to the ISF Facility Manager and are assigned responsibilities associated with the *Operator Training and Certification Plan.* The ISF Facility Manager's responsibilities include:

- Ensuring implementation of the Operator Training and Certification Plan
- Certifying in writing personnel authorized to supervise and/or operate equipment and controls important to safety.
- Designating in writing personnel authorized to act as Training Instructors or Subject Matter Experts. Those Training Instructors or Subject Matter Experts that are not Certified Operators shall have the basis for their qualification documented.

4.2 THE ADMINISTRATIVE SERVICES MANAGER

The Administrative Services Manager is responsible for ensuring the administration of the *Operator Training and Certification Plan.* These administrative duties include:

- Ensuring lesson plans, for topics identified in this training plan, are developed and approved.
- Ensuring required training is scheduled and conducted as required by this plan
- Ensuring records of training and qualification are completed and maintained as quality records.
- Providing periodic reports of qualification status to the ISF Operations Manager.

4.3 TRAINING INSTRUCTOR/SUBJECT MATTER EXPERT

Training Instructors and Subject Matter Experts report to the Administrative Services Manager and are responsible for implementation of training activities. These responsibilities include:

- Developing training material (i.e., lesson plans, OJT requirements, required reading)
- Conducting training using approved training material
- Developing and administering examinations (written and practical)
- Ensuring that required training records are completed and transmitted as quality records per the requirements of the QPP (Ref. 5)

4.4 OPERATIONS MANAGER

The Operations Manager is responsible for ensuring that only Certified Operators are assigned duties to supervise and/or operate equipment and controls important to safety. The Operations Manager also assists the Administrative Services Manager in identifying training needs based on reviews of operational performance. The Operations Manager's responsibilities include:

- Selecting personnel to become Certified Operators
- Identifying training weaknesses and recommending changes and/or enhancements to the *Operator Training and Certification Plan* and/or lesson plans and examinations
- Revoking certification of individuals based on performance deficiencies or other identified weaknesses. Revocation of certification will be documented in writing with notification to the ISF Facility Manager.
- Evaluating and documenting performance of Shift Supervisors at least annually.

4.5 SHIFT SUPERVISOR

The Shift Supervisor reports to the Operations Manager and is responsible for ensuring that only Certified Operators are assigned duties to direct and/or operate equipment and controls important to safety. The Shift Supervisor's responsibilities include:

- Providing on-shift supervision and training for personnel in the *Operator Training and Certification Plan*
- Evaluating and documenting performance of Certified Operators and Equipment Operators at least annually
- Recommending Equipment Operators for enrolled in the *Operator Training and Certification Plan*

5.0 REVISIONS

Changes to the Operator Training and Certification Plan may be made without prior NRC approval provided the change does not result in a decrease in the scope or effectiveness of the plan. The following are examples of items that are not considered to be a reduction in scope or effectiveness of the Operator Training and Certification Plan:

- Editorial changes to correct grammatical or spelling errors.
- Use of generic organizational position titles that clearly denote the position function, supplemented as necessary by descriptive text, rather than specific titles.
- Deletion of training requirements associated with systems, structures, or components that have been removed from ISF Facility service in accordance with the provisions of 10 CFR 72.48.
- Addition of training topics to cover facility modifications or administrative changes implemented in accordance with the provisions of 10 CFR 72.48.

Changes to the Operator Training and Certification Plan that do not reduce the scope or effectiveness of the plan will be submitted to the NRC biennially from the time of license receipt. Changes considered to be a reduction in scope or effectiveness will not be implemented without prior NRC approval.

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6.0 OPERATOR TRAINING AND CERTIFICATION PLAN

The Operator Training and Certification Plan includes an initial training course and biennial re-certification. Successful completion of this training plan is required for personnel that supervise and/or operate equipment and controls that have been identified as important to safety in the SAR.

6.1 INITIAL TRAINING

To become a Certified Operator an individual must complete the initial training course. The initial training course is comprised of academic and OJT.

6.1.1 Academic Training

The academic training phase of the Operator Training and Certification Plan shall consist, as a minimum, of lectures and/or self-study of topics in the following areas applicable to ISF Facility operations:

1. Thermodynamic and heat transfer theory
2. Nuclear decay heat generation
3. Nuclear criticality considerations
4. Fuel characteristics
5. Health Physics/Radiological Protection
   a) Types of radiation
   b) Principles of radiation protection
   c) As Low As Reasonably Achievable (ALARA) Concepts
   d) Time, distance, and shielding
   e) Personnel dosimetry
   f) Radiation detection
   g) Contamination control
   h) Radiation work procedures
   i) Protective clothing and respiratory protection
   j) Decontamination techniques
   k) 10 CFR Part 20, "Standards for Protection Against Radiation"
6. Facility layout and function
7. Equipment design features and operating characteristics of the following structures or components
   a) Transfer cask
   b) Transporter
   c) Cask receipt crane

d) Cask trolley

e) Transfer Tunnel doors

f) Fuel handling machine

g) Master slave manipulators

h) Decanning machine

i) Worktable

j) ISF canister/basket

k) Bench containment vessels

l) Fuel Packaging Area heating and ventilation system

m) Canister trolley

n) Canister Closure Area crane

o) Canister Closure Area welding machine

p) Vacuum drying system

q) Helium backfill system

r) ISF canister leak detection system

s) Canister handling machine

t) Storage vault

u) Storage tube

v) Effluent monitoring systems

w) Plant electrical distribution

x) Fire protection systems

y) Instrumentation and alarms

z) Waste handling systems

8. Facility requirements, procedures and limitations which, as a minimum, shall include:

a) ISF Facility Technical Specifications including the bases

b) Operating procedures for equipment and controls important to safety

c) Selected portions of 10 CFR 72

d) Selected administrative procedures

e) Selected maintenance procedures

f) Certified Operator responsibilities and authority

g) Emergency operating procedures

6.1.2 On-the-Job Training

OJT provides practical experience and the ability to evaluate personnel performance in an operating environment. OJT includes learning the basics of watch standing such as shift turnover, shift operations, and record keeping. Emphasis will be placed on the student's ability to follow procedures, evaluate conditions, and determine/perform appropriate remedial actions.

OJT involving actual and/or simulated performance shall be provided in the following areas:

1. Cask Receipt Area operations including:
   a) Transfer cask acceptance
   b) Movement of transfer cask from transporter to cask trolley
   c) Movement of transfer cask to Fuel Packaging Area
   d) Cask trolley operations
2. Fuel Packaging operations including:
   a) Unloading of transfer cask
   b) Operation of fuel handling machine
   c) Operation of master slave manipulators
   d) Operation of the decanning machine
   e) Operation of the worktable
   f) ISF basket and ISF canister loading operations
   g) Canister trolley operations
   h) Waste handling
3. Canister Closure Area operations including:
   a) Operation of welding equipment
   b) Operation of vacuum drying equipment
   c) Operation of helium backfill equipment
   d) Operation of leak detection equipment
4. Storage Area operations
   a) Canister handling machine operations
   b) Storage tube closure operations
   c) Inspection of vault inlet and outlet vents for blockage
   d) Storage tube interseal leak testing

OJT training performed prior to receipt of spent nuclear fuel (i.e., certification of initial operations staff) will be performed during pre-operational testing using dummy fuel assemblies and canisters. OJT during the initial operations staff training should simulate conditions, as close as possible, to those anticipated during operations. Upon completion of initial operations staff training, subsequent OJT will utilize actual operations, where practical.

6.1.3 Candidate Evaluation

This section discusses the evaluation process for becoming an Certified Operator. This section will discuss examinations including failures and provisions for training exemptions.

6.1.3.1 Examinations

Upon completion of the training plan, a comprehensive examination will be given to each Certified Operator candidate. The comprehensive examination shall include a written examination and a practical examination.

The written examination requires a minimum score of 80 percent to pass. The written examination is intended to provide an evaluation of the candidate's knowledge of facility design, theory of operation, as well as administrative and regulatory requirements. Examinations may be open reference (i.e., procedures and other plant documentation may be used).

The practical examination is intended to prove an evaluation of skills and abilities that cannot usually be evaluated in a written examination. Job performance measures, drills, and facility walk-throughs are the usual methods used for practical examinations. Practical examinations are graded on a pass/fail basis.

6.1.3.2 Examination Failures

An individual who fails to pass either the written or practical examination shall not perform the duties of an Certified Operator until he/she has completed a remedial training plan and passed an appropriate examination. Requirements for passing of re-examinations shall be the same as those for the initial examination. The ISF Facility Manager or designee will approve the remedial training plan and re-examination.

6.1.3.3 Exemption of Training Requirements

The ISF Facility Manager may exempt an individual from a specific training requirement based upon the individual's depth of experience and previous training. Such exemptions, including the basis, shall be documented. The requirement for a medical examination shall not be exempted.

6.1.4 Qualification

Certified Operators shall satisfy the following requirements.

- Complete the initial training course (Section 6.1).
- Score at least 80 percent on the comprehensive written examination.
- Pass the practical examination.

- Pass a medical examination by a physician to determine that the candidates medical condition could not cause operational errors that could endanger other ISF Facility personnel or the public health and safety as specified in 10 CFR 72.194.
- Be certified, in writing, by the ISF Facility Manager or designee.

6.2 RE-CERTIFICATION

The Certified Operator re-certification plan consists of lectures and/or self-study topics that covers selected areas presented during the initial training plan. The re-certification plan also includes provisions for continuing training to address facility design changes, program and procedure changes, and lessons learned from industry and ISF Facility operations experience reviews. Special emphasis will be placed on correcting weaknesses identified during annual performance reviews.

6.2.1 Re-Certification Schedule

The Certified Operator re-certification plan cycle shall span a period of 24 months. The cycle includes a practical and written examination.

6.2.2 Missed Training

Any missed training or examinations must be made up within 60 days. If the required training has not been completed, the qualification as Certified Operator shall be suspended pending completion of the missed training.

6.2.3 Re-Certification Evaluation

A comprehensive examination shall be administered as part of the re-certification cycle. The comprehensive examination shall include a written and practical examination.

The written examination requires a minimum score of 80 percent to pass. The written examination is intended to provide an evaluation of the candidate's knowledge of facility design, theory of operation, as well as administrative and regulatory requirements. Examinations may be open reference (i.e., procedures and other plant documentation may be used).

The practical examination is intended to provide an evaluation of skills and abilities that can not usually be evaluated in a written examination. Job performance measures, drills, and facility walk-throughs are the usual methods used for practical examinations. Practical examinations are graded on a pass/fail basis.

6.2.4 Examination Failures

An individual who fails to pass either the written or practical examination shall have their qualification as Certified Operator suspended until a remedial training plan has been completed and an appropriate examination has been passed. The ISF Facility Manager or designee shall approve the remedial training plan and re-examination.

6.2.5 Maintenance of Certified Operator Qualifications

To maintain qualification as a Certified Operator, the following requirements must be satisfied or exempted per Section 6.2.6.

- Complete the re-certification training required by Section 6.2.
- Score at least 80 percent on the biennial written examination.
- Pass the biennial practical examination.
- Pass a biennial medical examination by a physician to determine that the Certified Operator's medical condition could not cause operational errors that could endanger other ISF Facility personnel or the public health as specified in 10 CFR 72.194.

6.2.6 Exemption of Re-Certification Requirements

An individual may be exempted from specific re-certification requirements. Exemptions will be approved by the ISF Facility Manager and include a written basis for granting the exemption. Exemptions from re-certification requirements may be granted using the following criteria:

- Completion of similar training (e.g., completion of equipment-specific training provided by vendor or other SME).
- Completion of similar evaluation (e.g., completion of graded emergency plan drill that tested similar skills, abilities, or knowledge).
- Active participation in development, delivery, or evaluation of training (e.g., preparation of the biennial examination).

An individual shall not be exempted from two consecutive biennial practical or biennial written examinations. The requirement for a biennial medical examination shall not be exempted.

6.2.7 Plan Evaluation

Evaluations of the effectiveness of the re-certification plan shall be conducted during each 24-month re-certification cycle. The results of written and/or practical examinations shall be used to evaluate the effectiveness of re-certification. The results of these evaluations will be used to enhance the initial and re-certification plans.

7.0 TRAINING RECORDS

Training records for the Certified Operators will be maintained in accordance with the requirements specified in Section 9.4.2 of the SAR. As a minimum the following training records shall be maintained for at least 3 years:

- Results of each Certified Operator's biennial medical examination.
- Completed written examinations taken as part of the Operator Training and Certification Plan for each Certified Operator.
- Written documentation detailing the results of the practical examination (pass/fail) including strengths, weaknesses, and recommendations for additional training or retesting if warranted.
- The number of training hours and subjects covered for each operator (applicable to classroom training).
- Job performance records for each Certified Operator, including duties performed, the time spent at these duties, and an annual evaluation of job performance by the supervisor.
- Record of ISF Facility Manager certification for each Certified Operator.

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8.0 REFERENCES

1. Title 10, Code of Federal Regulations, Part 72, Subpart I, *Training and Certification of Personnel*, of 10 Code of Federal Regulations (CFR) 72, *Licensing Requirements for the Independent Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI).*

2. Nuclear Regulatory Commission Draft Regulatory Guide HF-608-4 dated March 1982.

3. Foster Wheeler Environmental Corporation (2001), *ISF Facility Safety Analysis Report*, ISF-FW-RPT-0033.

4. American Nuclear Society Selection, Qualification, and Training of Personnel for Nuclear Power Plants, ANSI/ANS-3.1-1993.

5. Foster Wheeler Environmental Corporation (2001), *Quality Program Plan* (QPP), ISF-FW-PLN-0017.

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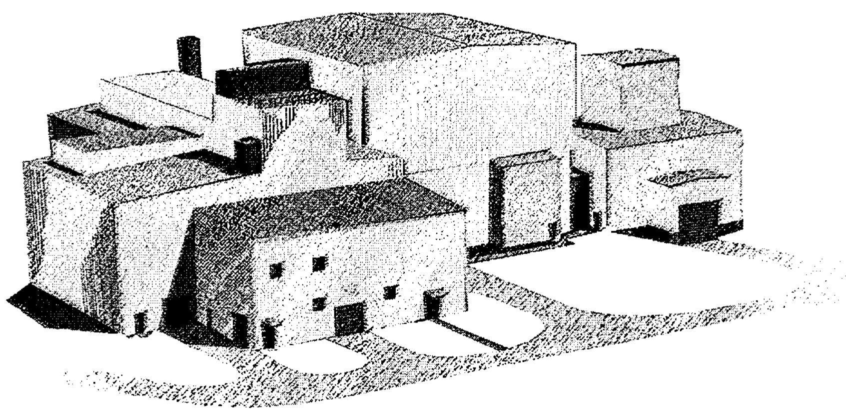
License Application

Appendix C

Proposed Decommissioning Plan

Idaho Spent Fuel Facility

Docket No. 72-25



ISF-FW-PLN-0027

FOSTER WHEELER ENVIRONMENTAL CORPORATION

Contents

1.0 GENERAL INFORMATION

1.1 INTRODUCTION

The Idaho Spent Fuel (ISF) Facility is adjacent to the Idaho Nuclear Technology and Engineering Center (INTEC) at the U.S. Department of Energy's (DOE) Idaho National Engineering and Environmental Laboratory (INEEL). The INEEL is one of nine multi-program laboratories in DOE's nationwide complex.

The INEEL occupies an 890-square-mile area of the Upper Snake River Plain in southern Idaho. The site measures approximately 37 miles north to south and 35 miles east to west. Most of the INEEL is within Butte County, but portions also extend into Bingham, Bonneville, Jefferson, and Clark counties.

The ISF Facility is designed, licensed, constructed, and operated by Foster Wheeler Environmental Corporation (FWENC), 1000 The American Road, Morris Plains, New Jersey, 07950.

The ISF Facility provides interim storage for spent nuclear fuel. In accordance with a settlement agreement between the DOE and the State of Idaho, the spent nuclear fuel must be removed from Idaho by 2035. It is anticipated that spent nuclear fuel will be transferred from the ISF Facility to the geologic repository, which is projected to become operational in 2010.

FWENC is under contract with DOE to operate the ISF Facility through 2010, after which the DOE will either extend its contract with FWENC, transfer the Nuclear Regulatory Commission (NRC) license for the facility to another contractor, or assume the license itself, after obtaining the necessary regulatory approvals. The DOE is contractually obligated to provide funding for decommissioning the facility (Ref. 1).

As discussed in the *Conceptual Plan for Decommissioning the INEL TMI-2 Independent Spent Fuel Storage Installation* (Ref. 2), waste acceptance criteria, documentation required under the National Environmental Policy Act (NEPA), regulatory requirements, and disposal characteristics of spent nuclear fuel have not been determined for the final disposition of DOE-owned spent nuclear fuel. Resolution of technical, regulatory, safety, legal, and institutional matters will be necessary before the spent nuclear fuel is moved from the INEEL.

This document is a conceptual plan for decommissioning the ISF Facility. Its objective is to demonstrate that the facility can be decommissioned in a manner that is both economical and safe. The plan describes the costs and activities required for safely removing the ISF Facility from service and reducing residual radioactivity through remediation to a level that permits release of the property and termination of the NRC license (Ref. 3). The plan is designed to allow flexibility in the decommissioning activities that are actually implemented so new technology can be incorporated when appropriate.

The primary areas of anticipated radioactive contamination at the ISF Facility are the Transfer Area, Solid Waste Processing Area, heating, ventilation, and air conditioning (HVAC) systems, and the portions of systems that contained radioactive fluids. Since the exterior of the storage canisters will not come into contact with radioactive materials, the canisters should not become contaminated. After the canisters are removed from the ISF Facility Site, the Storage Area should therefore require little or no remediation.

Limiting the number of areas that will require remediation increases the likelihood that this plan can be implemented safely and economically.

1.2 CONTENTS OF THE PROPOSED DECOMMISSIONING PLAN

This proposed decommissioning plan was prepared in accordance with NRC Regulatory Guide 3.65, which applies to final decommissioning plans. Accordingly, certain information contained herein relates to activities that are prospective in nature, and will be subject to revision based on knowledge gained over the course of facility operation. The plan is intended to provide assurance that the ISF Facility will be safely and efficiently decommissioned by examining the elements that will make up the final decommissioning plan. The plan discusses decommissioning methodology and organization, estimated costs, major tasks and schedules, and protection of occupational and public health and safety, including site characterization, radiation protection, waste management, and physical protection provisions.

A final decommissioning plan will be submitted prior to the termination of the NRC license, in accordance with the requirements of 10 CFR 72.54 (Ref. 4). At a minimum, the final plan will include:

- description of current condition of the ISF Facility
- choice of decommissioning alternative to be implemented
- description of controls and limits on procedures and equipment
- description of final survey
- updated cost estimate
- description of technical specifications and quality assurance provisions

2.0 DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

The three decommissioning alternatives described in NUREG-0586 (Ref. 5) are as follows:

- DECON – equipment, structures, and other portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the termination of the license shortly after cessation of facility operations.
- SAFSTOR – the facility is placed and maintained in a condition that allows it to be safely stored and subsequently decontaminated to levels that permit the termination of the license.
- ENTOMB – radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained, and continued surveillance is carried out until the radioactivity decays to a level that permits the termination of the license.

The decision concerning which alternative to implement will be made during the decommissioning planning phase. The decision will be based on many factors, including:

- physical condition of equipment and structures over a long-term period
- optimization of radiological aspects
- environmental impacts of the project
- existence of technical resources
- availability of waste management and disposal facilities
- costs
- public opinion

To minimize the impacts on both the environment and the community, it was decided that the best alternative for presentation in the conceptual decommissioning plan is DECON. As such, the proposed approach to decommissioning the ISF Facility is to decontaminate equipment and building surfaces, demolish and completely remove the building, and free release as many items as possible for recycling/salvage. The design of the facility, the selected construction materials, and the aggressive preventive and protective methods used during the operating life (i.e., "start clean, stay clean" concept) will minimize the amount of actual decontamination required during decommissioning. Therefore, a majority of building surfaces and some equipment should be released for unrestricted use.

Equipment and surface decontamination methods have been chosen to minimize secondary wastes and ensure the maximum amount of free-releasable items without unnecessarily inflating costs. By operating under the "start clean, stay clean" concept, the disposal of large amounts of radiologically contaminated materials at the end of plant life is avoided. This is important because wholesale disposal places an unnecessary burden on the nation's waste handling system and increases the potential for the public to be exposed to radiologically contaminated wastes.

2.1 DECOMMISSIONING OBJECTIVE, ACTIVITIES, TASKS, AND SCHEDULES

2.1.1 Decommissioning Objective, Activities, and Tasks

2.1.1.1 Introduction

Planning for decommissioning the ISF Facility began in the design phase using As Low As Reasonably Achievable (ALARA) principles and incorporating specific design features that would facilitate decommissioning. These design features focus on reducing the residual radioactive inventory, the time required to perform decommissioning tasks, the time personnel must spend in high-contamination areas, and the generation of radioactive waste.

ISF Facility features that improve the decommissioning process include:

- compartmentalizing the various ISF Facility processes, including maximizing the amount of support equipment located outside the radiological control area (RCA)
- applying protective coatings on concrete and steel surfaces in areas that may become contaminated
- providing ready access to the liquid storage tank
- minimizing the amount of potentially contaminated equipment directly embedded in the concrete floors, walls, or ceilings (e.g., piping)
- minimizing the amount of piping inside tanks

Decommissioning activities will begin with the cleanout of the Fuel Packaging Area. Systems will be vacuumed or flushed, as appropriate, to remove any residual materials, and contaminated filters will be removed from equipment for safe disposal. These techniques will reduce worker exposure.

As required by facility operation procedures, a complete history of materials processed through the Transfer Area and facility maintenance activities will be maintained along with accounts of spills and clean-up actions. This historical record will be available for making needed revisions to the decommissioning plan before final decommissioning operations begin.

Written maintenance records, including complete descriptions of spills and operational records will be maintained for the decommissioning and will lead to more efficient practices. Characterization will be easier since the history of spills, equipment replacements, and facility maintenance will be known. Shortly before the conclusion of the operations, these records will be reviewed and the information will be incorporated into the final decommissioning plan. Decommissioning operations will then commence.

Decommissioning of the ISF Facility is divided into two broad phases: (1) decontamination and dismantling and (2) site restoration.

The decontamination and dismantling phase will begin after all spent nuclear fuel has been transferred from the ISF Facility to the geologic repository. Major activities that will occur during this phase include:

- removing contaminated systems and components
- decontaminating structures
- performing final radiation survey

The Site Restoration phase will begin immediately after the decontamination and dismantling phase is completed, although some site restoration activities may occur during the decontamination and dismantling phase. The Site Restoration phase will involve the final disposition of structures, systems, and components. Systems, components, and structures required to contain and control radioactive materials during decommissioning activities will be identified and excluded from any restoration until no longer required. These excluded systems will then be decontaminated and removed for the performance of the final site survey. Site restoration activities not involving radioactive materials may be completed following the termination of the NRC license.

2.1.2 Decontamination and Dismantlement

Decontamination and dismantling phase activities for ISF Facility decommissioning involve the reduction of radioactivity to acceptable levels, allowing the termination of the NRC license. For the ISF Facility, these activities should be limited because of the way the facility and processes are designed. The anticipated areas of contamination are the Fuel Packaging Area, Solid Waste Processing Area, HVAC systems, and the portions of systems that contained radioactive liquids.

During this phase, contaminated systems and components will be handled in one of two ways: (1) they will be decontaminated and removed or (2) they will be removed, packaged, and then either shipped to an offsite processing facility or low-level radioactive waste disposal facility. HVAC systems will be removed in such a manner to ensure airflow is maintained from the least contaminated areas to areas of higher contamination.

Decontamination of facility structures may be completed concurrently with equipment removal. A comprehensive radiation survey will be completed following the removal or decontamination of contaminated systems, components, and structures.

2.1.2.1 Decontamination Methods

Contaminated systems and components will typically be decontaminated, removed, and released. They can also be sent to an offsite processing facility or a low-level radioactive waste disposal facility.

A variety of techniques may be used to decontaminate structures, ranging from washing with water to removing surface material. Contaminated structural material will be handled in the same manner as contaminated systems and components. Although large-scale chemical decontamination is not anticipated as part of the decommissioning, small applications of chemicals may be used on systems or tanks to reduce radiation dose rates prior to dismantling or decontaminating general areas. Other typical decontamination methods include wiping, washing, vacuuming, scabbling, spalling, and abrasive blasting. Selection of the preferred decontamination method will be based on the specific situation. The decommissioning cost estimate was based on scabbling.

Coatings may be applied and hand wiping may be used to stabilize or remove loose surface contamination. Airborne contamination control and waste processing systems will be used as necessary to control and monitor releases. If structural surfaces are washed to remove contamination, controls will be established to ensure that wastewater is collected in liquid waste processing systems.

Tanks and vessels will be evaluated and, if required, flushed or cleaned prior to sectioning and/or removal to reduce contamination levels and remove sludge. The following considerations will be incorporated into sludge removal activities for tanks and vessels:

- Precautions will be taken to ensure that liquid inadvertently discharged from the tank is captured.
- Sludge removed from the tank will be stabilized prior to shipment.
- Wastewater will be processed and analyzed before being discharged.

Contaminated concrete may be removed and sent to a low-level radioactive waste disposal facility or disposed of by other methods in accordance with applicable regulations. Concrete will be removed using methods that control the removal depth to minimize the volume of waste. Vacuum removal of the dust and debris with high efficiency particulate air (HEPA) filtration of the effluent will be used, as applicable, to minimize the spread of contamination and reliance on respiratory protection measures.

It is anticipated that no more than 0.5 inch of concrete will need to be removed from the concrete surfaces inside the Fuel Packaging Area; this concrete will be handled as low-level radioactive waste as will the port plugs. The remaining concrete will be surveyed as necessary and disposed of as construction debris.

2.1.2.2 Dismantlement Methods

Two types of dismantlement methods – disassembly and cutting or other destructive methods – are likely to be used, although other appropriate technologies could be used. Selection of the preferred dismantlement method will depend on the specific situation.

Disassembly generally consists of removing fasteners and components in an orderly, nondestructive manner (i.e., the reverse of the steps used during original assembly). Cutting methods include flame cutting, abrasive cutting, and cold cutting.

Flame cutting is performed using oxy-acetylene and other gas torches, carbon arc torches, air or oxy arc torches, plasma arc torches, and cutting electrodes, or combinations of these. Abrasive cutting is performed using grinders, abrasive saw blades, wire saws, water lasers, grit blast, and other techniques that wear away metal. Cold cutting includes the use of bandsaws, bladesaws, shears, and metal cutters.

Concrete demolition techniques may include use of impact hammers, grapples, or other standard demolition techniques.

System dismantling will include removing valves and piping for disposal. Most valves can be removed along with the piping. Larger valves and valves with actuators may be removed separately. Valve actuators that can be decontaminated will be removed from the valves prior to pipe removal where practical.

2.1.3 Procedures

Procedures used during construction and operation of the ISF Facility may also be used during decommissioning. Any changes to these procedures would comply with 10 CFR 72.48 (Ref. 4).

2.1.4 Scheduling

A preliminary decommissioning 24-month schedule was developed to support this proposed decommissioning plan. During the decommissioning planning phase, a final decommissioning schedule will be created. The sequence of decommissioning activities may be dictated by access and material handling restrictions or by personnel exposure considerations. All work activities will be planned to minimize the spread of contamination.

In most parts of the facility, uncontaminated or only slightly contaminated items will be removed first to avoid contaminating or further contaminating them when more highly contaminated equipment is removed. However, personnel exposure considerations may not always allow this option. When uncontaminated equipment cannot be removed first, covers or other protection will be used to minimize the spread of contamination. Similarly, uncontaminated piping will generally be removed from pipe chases and horizontal pipeways before contaminated pipes are cut. If this is not possible, covers or other protection will be used.

Where rapid cutting techniques are appropriate, pipes and equipment can be reduced in place to pieces that are of manageable size using light rigging or manual lifting. Where slow cutting techniques are used, the largest manageable pieces will typically be freed and their sizes reduced at a more convenient location.

The ISF Facility will be equipped with cranes, hoists, forklifts, and lifting and transport systems. These systems will be used to lift and transport components and equipment to support decommissioning activities. Installed cranes, hoists, and other lifting devices will be decontaminated and dismantled when they are no longer needed to support decommissioning activities.

In areas where a considerable amount of material movement is expected, such as in the pipe penetration areas and pipe chases, hoisting equipment (e.g., winches and hoists) may be attached to the building's structural steel. Beam clamps and welded lugs on the steel will allow repositioning of hoisting lines throughout an area.

In some areas of the facility, it may be convenient to use material handling equipment, such as forklifts or front-end loaders, for moving materials from one location to another. Small mobile cranes can be used inside facility structures for smaller equipment and materials. Wheeled carts can be used for moving pipe, steel, and other items. Skid rails, skid ways, and air pallets may be used for moving larger equipment.

To minimize the potential for the spread of contamination, the following considerations will be incorporated into the planning of decommissioning activities:

- Contaminated liquids will be contained within existing or supplemental barriers and processed prior to release.
- Isolation of electrical and pneumatic systems from components prior to their dismantlement.

- Covering of openings in internally contaminated components to confine internal contamination.
- Removal of contaminated supports in conjunction with equipment removal or decontamination of supports in conjunction with decontamination of the building.
- Removal of contaminated systems and components from areas and buildings prior to structural decontamination. Block shield walls or portions of other walls, ceilings, or floors may be removed to permit removal of systems and components.
- Removal or decontamination of embedded contaminated piping, conduit, ducts, plates, channels, anchors, sumps, and sleeves during structural decontamination activities in buildings and facility areas.
- Establishing local or centralized processing and cutting stations to facilitate packaging of components removed in large pieces.
- Removal of intact components if they are small or compact, including most valves, smaller pumps, some tanks, and heat exchangers. These components could then be fully or partially decontaminated and, if necessary, reduced to smaller dimensions in preparation for disposal or release.

2.2 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

At the time the final decommissioning plan is prepared, information will be provided concerning personnel who will be responsible for, or participate in, decommissioning activities. This information will include:

- decommissioning staff positions and responsibilities
- minimum qualifications for these positions

Experienced and knowledgeable personnel will perform the technical and administrative tasks required during decommissioning. To the extent practicable, the decommissioning team will include personnel previously employed at the ISF Facility to capitalize on their familiarity with the facility. However, contractors may be used to provide specialized services or to supplement the facility staff.

2.3 TRAINING PROGRAM

The ISF Facility training program is designed to provide instruction to ensure that personnel have the knowledge and skills necessary to perform their job functions safely. Training applicable to specific activities, tasks, and conditions will be developed or discontinued as decommissioning progresses. The training program will be maintained throughout decommissioning as necessary to provide the ISF Facility personnel with the specialized training and technical skills necessary to maintain the facility in a safe condition.

2.3.1 General Training

Persons requiring access to the ISF Facility will receive general training, which will include the following representative topics:

- introduction to the ISF Facility
- fundamentals of radiological protection
- techniques to maintain radiation exposure ALARA
- emergency response plan
- facility safety
- fire protection
- chemical safety
- physical protection
- quality assurance

The content of the course may be revised, as needed, during decommissioning.

2.3.2 Task-Specific Training

Task-specific training for selected activities will include the appropriate level of training in decontamination, decommissioning activities, and radiation protection. Managers will ensure that employees and contractors who perform decommissioning activities are properly trained, qualified, and proficient in the principles and techniques of activities necessary to perform their assigned tasks, in accordance with approved procedures.

2.3.3 Training Records

Records of training will be maintained in accordance with the ISF Facility's records management program.

2.3.4 Instructor Qualification

The background, qualifications, and experience of instructors will be appropriate for the subject matter. Instructor qualifications will be administratively controlled by approved procedures.

2.4 CONTRACTOR ASSISTANCE

2.4.1 Contractor Scope of Work

During decommissioning, contractors may be used to provide specialized services or to supplement facility personnel. Tasks for which contractors may be used to provide support during decommissioning may include, but are not limited to, the following:

- processing, packaging, transportation, and disposal of radioactive material
- decontamination and recycling of radioactively contaminated material
- radiation protection
- fire protection
- design and fabrication of special dismantling equipment
- engineering and design services, such as heavy loads management and transportation engineering
- dismantlement and demolition of components, systems, and structures

2.4.2 Contractor Administrative Controls

During decommissioning planning, the responsibility for contractor control, including the contractor's effectiveness in performing to bid specifications, will be identified. The responsible entity will provide management oversight to ensure that tasks performed by the contractors are in full compliance with the Quality Program Plan for the ISF Facility, the purchase agreement, and applicable regulatory requirements.

2.4.3 Contractor Qualifications and Experience

Potential contractors for decommissioning activities will be required to supply their qualifications as part of bid specifications. These qualifications will be evaluated and reviewed for:

- safety record
- safety program
- demonstrated experience in providing services on similar projects
- cost and schedule compliance
- technical and operational capability
- ability to meet regulatory requirements
- financial reliability
- evaluation of key personnel qualifications

3.0 DESCRIPTION OF METHODS USED FOR PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

3.1 FACILITY RADIOLOGICAL HISTORY AND STATUS

During decommissioning planning, the historical information related to operation of the ISF Facility as well as information related to work with radioactive materials will be consolidated. This information will be used to develop the plans for ISF Facility site decontamination.

As required by facility operation procedures, a complete history of materials processed through the Transfer Area, facility maintenance activities, and accounts of spills and clean-up actions will be recorded and will be available to use in revising this plan prior to starting ISF Facility decommissioning operations.

Written maintenance records, including descriptions of spills, and operational records, will be maintained for the decommissioning. Site characterization will be easier since the history of spills, equipment replacements, and facility maintenance will be known. Shortly before the conclusion of the operations, these records will be reviewed and the information will be incorporated into the final Decommissioning Plan. After the processes have been safely shutdown, the decommissioning plan will be updated to reflect the current conditions, and decommissioning operations will commence.

ISF Facility site characterization is an ongoing process similar to that described in NUREG/CR-5849 (Ref. 6). The ISF Facility site characterization can be completed in three phases:

- Phase I – scoping survey/site characterization
- Phase II – radiological surveys to support ISF Facility decontamination and dismantlement
- Phase III – final radiation survey

Phase I will be completed during the decommissioning planning phase, which will occur at least 2 years prior to the anticipated termination of the license. This information will be necessary to support revised decommissioning cost estimates and decision making, to determine the location and extent of contamination, and to collect background information to help facilitate the termination of the license. Phase II involves routine radiological surveys to support facility decontamination and dismantlement. Phase III consists of the final radiation survey, which will demonstrate ISF Facility radiological conditions are within license termination criteria.

3.1.1 Structures

Operational radiation protection survey data will be supplemented by data from additional surveys to determine the presence and/or level of contamination in structures. Structures with known contamination will be surveyed to characterize the extent of contamination.

As described in NUREG/CR-5849 (Ref. 6) and NUREG-1575 (Ref. 3), the radiological history of the ISF Facility will be used to select biased sampling locations for potentially contaminated areas as part of the site characterization survey.

Decontamination in areas where the contamination is removable will be performed by simple methods, such as wiping or mopping. Fixed contamination can be removed using surface destruction techniques (e.g., scabbling). Additional surveys for fixed contamination will be performed as radiation levels are reduced by removal of radiation sources.

3.1.2 Systems

Each facility system will be evaluated to determine the likelihood that it is contaminated and will also be sampled by direct surveying, loose surface swiping, or metal scrapings. Systems will be surveyed to estimate the quantity of contamination. Detected activity that cannot be identified as naturally occurring will be attributed to facility operations, and the system will be classified as contaminated. The approach will involve grouping facility systems into four categories: contaminated, potentially contaminated, indeterminate (need more data), and not contaminated.

If a system is sufficiently contaminated, its curie content may be estimated by measuring the dose rate and multiplying by the total contaminated surface area of the system to conservatively estimate the curie content. To determine radionuclide spectrum, system scrapings may be taken.

For systems with low or slight contamination, it may not be possible to approximate activity deposition by field dose-rate measurements. Scrapings from these systems will be used to determine activity deposition and curie content.

Swipes will be taken to measure removable surface contamination, and scrapings will be collected to determine fixed-surface contamination. Individual system scrapings will be analyzed to determine a qualitative radionuclide spectrum.

The total system burial volume is estimated to weigh approximately 28,000 tons. The volume estimates are based on material take-offs and unit pricing.

3.1.3 Activation

Facility storage tubes in the Storage Area are not anticipated to become significantly activated during the storage period of the spent nuclear fuel.

3.1.4 Environment

The environmental survey, which will include representative outdoor areas, will focus on the impacts of ISF Facility operation on the environment. Operational and preoperational environmental monitoring data will be used to measure and evaluate the impacts. Additional sampling will be conducted to augment, or better define, areas requiring surveys. Survey results will be compared to background data to determine the overall consequences of ISF Facility operation.

3.2 RADIATION PROTECTION AND ALARA PROGRAM POLICIES

Radiation protection policies are expected to be followed in activities and decisions related to radiation protection and radiological controls. The policies and requirements of the radiation protection program are described in the radiological and contamination control program.

The ALARA measures incorporated into the design of the facility to ensure the safety of the personnel during operation will also provide a measure of protection during the decommissioning phase (shielding will be used, etc.). Complete training of the decommissioning personnel as well as thorough planning for each decommissioning activity will also contribute to achieving compliance with ALARA principles.

In order to reduce general area dose rates in the building, especially in the Fuel Packaging Area and the Solid Waste Processing Area, routine decontamination will be performed during operations. Equipment and building surfaces will be decontaminated and/or surface contaminants will be fixed prior to the commencement of the decommissioning phase.

Total radiation exposures to facility personnel and the public must be maintained ALARA to be in compliance with the regulatory requirements of 10 CFR 20 (Ref. 7) and 10 CFR 72 (Ref. 4). Radiological hazards will be monitored and evaluated on a routine basis to maintain radiation exposures ALARA. Radiation protection training will be provided to occupationally exposed personnel to ensure that they understand their responsibility to follow procedures and to maintain their exposure to radiation ALARA.

3.3 RADIATION PROTECTION PROGRAM

3.3.1 Radiation Protection Organization

Under the DECON alternative, the radiation protection staff would be augmented with additional personnel, if necessary.

3.3.2 Management Responsibilities

The ISF Facility decommissioning management shall establish specific ALARA goals and objectives for the radiation protection program and ensure that work specifications, designs, and work packages involving radiation exposure or handling of radioactive materials incorporate effective radiological controls. Implementation of specific ALARA actions and their incorporation into daily work activities will be the responsibility of each individual manager, supervisor, and worker.

3.3.3 Radiation Protection Program Implementation

The radiation protection program will be implemented, maintained, and audited in accordance with approved facility procedures that establish controls for equipment, instrumentation, and monitoring.

Implementing procedures for facility and radiation protection will direct the use of various practices and equipment to ensure that general facility area contamination is controlled at the source to the greatest extent possible. Additional contamination controls will be specified for jobs involving high levels of contamination (e.g., a double step-off pad, additional surveys). Appropriate contamination controls will be used when contaminated tools and equipment are carried between areas. Radiation monitoring equipment will be located in the facility so that personnel can determine if they have been contaminated prior to entering another area of the facility. The final checkpoint for personnel leaving controlled areas of the facility is the access control point. Temporary exit points may be established at remote control areas, as needed.

Radiation protection personnel will perform routine radiation surveys of accessible areas of the facility. These surveys will consist of contamination surveys, air sample collection, and external radiation measurements as appropriate for the specific area. Additionally, specific surveys will be performed as needed for operational and maintenance functions involving potential exposure of personnel to radiation or radioactive materials.

Personnel radiation exposure will be maintained ALARA using a combination of shielding, access control, contamination control, work planning, training, and sound radiation protection practices implemented through ISF Facility procedures.

Implementation of the ALARA program will include the frequent communication of ALARA actions and progress achieved towards ALARA goals to the management of the facility. Such communication will serve to raise ALARA awareness among the managers and promote teamwork toward achieving ALARA goals.

3.4 CONTRACTOR PERSONNEL

The ISF Facility training program is provided to personnel, including contractors, to ensure that individuals have adequate knowledge and skills to perform their job functions safely in a radiological environment.

3.5 RADIOACTIVE WASTE MANAGEMENT

Radioactive waste management activities during the ISF Facility decommissioning include activities related to the processing and disposal of liquid and solid radioactive waste.

The processing and disposal of liquid and solid radioactive waste will be managed in accordance with the radiation protection program, radioactive effluent controls program, and radiological environmental monitoring program.

The ISF Facility policy for control of radioactive wastes is to minimize the amount of waste material generated and to maintain the discharge of radioactive material at levels below the design objectives. To ensure that waste minimization goals are achieved during decommissioning, radiation workers will receive training in waste minimization procedures and practices.

3.5.1 Radioactive Waste Processing

Airborne radioactive particulates will be filtered through HEPA filters in the ventilation system for the buildings, portions of which will continue to be operated during decontamination and dismantlement. Temporary local ventilation systems with HEPA filtration, or other approved systems, may be used instead of, or to supplement, building ventilation for activities expected to result in the generation of airborne radioactive particulates.

Portions of the liquid radioactive waste system will continue to operate during decommissioning to process liquid radioactive wastes. Temporary liquid waste processing systems may also be used to process liquid radioactive waste.

Solid radioactive waste generated during decommissioning will be processed in accordance with facility procedures and sent to an offsite disposal facility.

Mixed waste is waste that contains both a hazardous waste component regulated under Subtitle C of the Resource Conservation and Recovery Act and a radioactive component consisting of source, special nuclear, or byproduct material regulated under the Atomic Energy Act. Facility procedures provide guidance for the minimization, control, and storage of mixed waste in accordance with Environmental Protection Agency (EPA) and NRC regulations. The use of potentially hazardous materials in radiological controlled areas will be minimized by ISF Facility procedures.

3.5.2 Radioactive Waste Disposal

Packaging, storage, and shipment of radioactive waste generated during decommissioning will be controlled by facility procedures. Facility procedures include requirements for:

- sorting and segregating radioactive waste and processing it to an acceptable form
- classifying radioactive waste in accordance with U.S. Department of Transportation (DOT) and NRC requirements
- packaging and labeling radioactive waste in accordance with DOT and disposal site criteria
- storing radioactive waste
- shipping radioactive waste in accordance with DOT and NRC requirements

3.5.3 Waste Disposition Categories, Quantities, and Disposition

The preliminary waste disposition categories/plan, waste volumes, and decommissioning cost estimates presented in this proposed decommissioning plan are based on the assumptions listed below, which will be confirmed during site characterization activities.

The cost estimate does not include the cost of the decontamination activities that will take place during the operation of the ISF Facility; that cost is included in the operations costs. Decontamination of areas before servicing is captured in the maintenance costs.

The cost estimate includes the cost of gross decontamination (terminal clean out) of contaminated areas prior to the commencement of dismantlement.

Concrete

- It is anticipated that no more than 0.5 inch of concrete will need to be removed from the concrete surfaces inside the Fuel Packaging Area, FHM, and Waste Processing Area and managed as low-level radioactive waste.
- The port plugs and hoist well plug will be managed as low-level radioactive waste.
- The remainder of the concrete and embedded rebar will be provided with enough protection by protective coatings and preventive decontamination measures used during operations that it can be decontaminated using standard wiping, washing, and vacuuming during decommissioning activities. This concrete will be disposed of as construction debris.

Structural Steel

- Except for steel inside the Fuel Packaging Area, structural steel will be decontaminated if necessary and then handled as construction debris.
- Steel inside the Fuel Packaging Area will be handled as low-level radioactive waste and recycled as radioactive scrap (e.g., metal melt process or volume reduction).

Equipment

- The liquid radioactive waste storage tanks will be recycled as radioactive scrap
- Piping will be flushed, dismantled, and disposed of as low-level radioactive waste
- Equipment inside the Fuel Packaging Area will receive gross decontamination to reduce the dose, and be disposed of as low-level radioactive waste.
- Equipment inside the solid waste handling area will receive gross decontamination to reduce the dose, and be disposed of as low-level radioactive waste.
- Exhaust components for heating, ventilation, and air-conditioning (HVAC) systems upstream of the final HEPA filters and supply components for HVAC systems downstream of the local HEPA filters and the HEPA filters will be packaged as low-level radioactive waste. The use of efficient packaging methods at the ISF Facility will reduce the volume of this waste; other methods of volume reduction will be evaluated during revisions to the decommissioning plan.
- All other equipment will be surveyed as necessary, and if free released, sold as clean scrap or disposed of as construction debris.

Electrical Items and Equipment

- Electrical items (e.g., cable, conduit, fittings) and equipment (e.g., motors) inside the Fuel Packaging Area will be handled as low-level radioactive waste.
- Other electrical items and equipment will be sold as clean scrap or disposed of as construction debris.

Special Items

- Forklift batteries are not expected to be classified as radioactive waste but will be handled on a special-case basis.
- Uninterruptible power supply (UPS) batteries are not expected to be classified as radioactive waste but will be handled on a special-case basis.
- Miscellaneous contaminated liquids (including hydraulic fluids, oils, and grease) may be adsorbed and disposed of as low-level radioactive waste
- Hydraulic fluids, oils, and grease will be disposed of in a commercial waste incinerator.

It is anticipated that the waste and debris generated from the decommissioning of the ISF Facility will fall into the categories defined and discussed below.

Category A-1. Heavy Non-Contaminated Metals

Category A-1 consists of metals that are not contaminated or can readily be decontaminated to free-release levels. Examples from the ISF Facility are structural steel members and larger pieces of equipment.

There will be an estimated 1175 tons of Category A-1 materials that can be free-released for reuse or recycling.

Category A-2. Heavy Contaminated Metals

Category A-2 includes process equipment and dense steel items that may be suitable for recycling as radioactive scrap.

There will be an estimated 309 tons of Category A-2 materials that may be recycled as radioactive scrap.

Category B-1. Non-Contaminated Metals

Category B-1 includes most of the process equipment and supporting services, such as electrical components and parts of the HVAC systems. Although much of this equipment will not be contaminated, it would not be economical to reuse or recycle it. This material will be disposed of as construction debris.

There will be an estimated 242 tons of Category B-1 materials that will be disposed of in a construction debris landfill.

Category B-2. Contaminated Metals

Category B-2 consists of contaminated process-related equipment. This equipment will be packaged and shipped for disposal as low-level radioactive waste because it probably would not be economical to decontaminate and survey it for free release. The use of efficient packaging methods at the ISF Facility will reduce the volume of this waste; other methods of volume reduction will be evaluated during revisions to the decommissioning pan

There will be an estimated 30 tons of Category B-2 materials that will be disposed of as low-level radioactive waste.

Category C-1. Non-Contaminated Concrete

Category C-1 includes the structural concrete and rebar at the ISF Facility. These materials will not be contaminated and will be disposed of in a construction debris landfill.

There will be an estimated 24,420 tons of Category C-1 materials that will be disposed of in a construction debris landfill.

Category C-2. Contaminated Concrete

Category C-2 includes the 0.5-inch-thick layer that will be removed from concrete surfaces in the Waste Processing Area and the Fuel Packaging Area. It also includes the Fuel Packaging Area port plugs and hoist well plug. This material will be disposed of as low-level radioactive waste.

There will be an estimated 112 tons of Category C-2 materials that will be disposed of as low-level radioactive waste.

Category D-1. Non-Contaminated Miscellaneous Materials

Category D-1 includes a variety of materials, such as miscellaneous cardboard, paper, plastic, wood, strapping, non-asbestos insulating materials, plumbing fixtures, flex connections, interior doors, window panes/glass, polyethylene, non-asbestos personnel protective equipment (PPE), non-asbestos prefilters, empty material containers, non-asbestos water filters, polyvinyl chloride (PVC) piping, acoustic ceiling, gypsum material, partitions, and incandescent bulbs (lamps). There will be approximately 14 tons of Category D-2 materials that will be disposed of as construction debris.

Category D-2. Contaminated Miscellaneous Materials

Category D-2 includes a variety of contaminated materials, such as miscellaneous cardboard, paper, plastic, wood, strapping, non-asbestos insulating materials, plumbing fixtures, flex connections, interior doors, window panes/glass, polyethylene, non-asbestos PPE, non-asbestos prefilters, empty material containers, non-asbestos water filters, PVC piping, acoustic ceiling, gypsum material, partitions, and incandescent bulbs (lamps). The largest component in this category will be the strippable coatings removed from non-concrete building and equipment surfaces. Stripped paint coating will be contaminated (its function is to remove contamination) and will be disposed of as low-level radioactive waste.

There will be an estimated 36 tons of Category D-2 materials that will be disposed of as low-level radioactive waste.

Category E. Special Materials

Category E materials contain potentially hazardous wastes; these materials include paint remover, hydraulic fluid/oil/grease, acid, and fluorescent bulbs. Category E materials will be collected and disposed of in an appropriate manner as radioactive waste, mixed waste, or construction debris.

It is estimated that only a small quantity, less than one cubic meter, of this waste will be generated during the decommissioning phase. For estimation purpose, one ton of Category E low-level radioactive waste was considered.

Category Other

These materials include 10,000 gallons of decontamination solution that will be disposed of as low-level radioactive waste and nearly 2000 tons of concrete and rebar from the support facility that can be disposed of in a construction waste landfill.

4.0 PROPOSED FINAL RADIATION SURVEY PLAN

4.1 INTRODUCTION

A final radiological survey will be performed to determine the condition of the ISF Facility site after decontamination activities have been completed. The purpose of the survey is to demonstrate that radiological conditions at the site meet license termination criteria. A detailed plan for the survey will be submitted to the NRC for approval prior to the final survey and the submittal of the application for license termination.

This section provides a brief overview of the methodology and criteria that will be used to develop the final survey plan. Guidance for developing the plan will be obtained from NUREG-1575 and NUREG/CR-5849 (Ref. 3 and 6). Reference 3 provides statistical approaches to survey design and data interpretation used by the EPA. Survey methods discussed in References 3 and 7 use state-of-the-art, commercially available instrumentation for conducting radiological surveys for decommissioning.

The final survey results will be provided to the NRC to support license termination. The final survey will be designed so that the NRC can verify procedures, results, and interpretations.

4.2 FINAL RELEASE CRITERIA

Criteria for contamination, exposure, and concentration levels are designed to ensure that radioactivity at the site is reduced to levels that allow termination of the license. Release of the site, facility, and materials will be based on release criteria for surface contamination, direct exposure, and soil and water concentrations consistent with 10 CFR 20, Subpart E, *Radiological Criteria for License Termination.* NUREG-1500 (Ref. 8) provides additional guidance for site release criteria.

4.2.1 Limits for Loose and Fixed Surface Contamination

Regulatory Guide 1.86 (Ref. 9) will be used to establish criteria for the release of materials, equipment, and structures with loose and fixed surface contamination for unrestricted use.

4.2.2 Limits for Direct Exposure

NUREG-0586 (Ref. 5) specifies a limit of 5 mrem/hour above background for direct exposure from residual radioactivity.

4.2.3 Limits for Total Concentrations in Soil and Water

NUREG-1500 (Ref. 8) provides generic dose conversion factors to derive the potential total effective dose equivalent (TEDE) to an average individual among those that could potentially receive the greatest exposure from residual radioactivity. The TEDE to the average individual will be maintained below the established limits for the generic pathways based on the methods described in Reference 9.

4.2.4 Limits for Unrestricted Release of Material

Equipment and materials from the ISF Facility will be surveyed for loose and fixed contamination prior to their removal from the site. Consistent with approved facility procedures, equipment and materials with less-than-detectable radioactive contamination limits will be unconditionally released. Contaminated equipment and materials that cannot be decontaminated will be treated as radioactive waste. Contaminated waste will be packaged and shipped to a low-level radioactive waste burial site or to a radioactive waste volume reduction facility prior to ultimate disposal at a burial site. Alternative disposal methods may also be considered.

4.3 PLANNING AND DESIGNING THE FINAL SURVEY

A final survey to determine the condition of the site will be performed after decontamination activities are completed to demonstrate that radiological conditions satisfy the final release criteria. The final survey results will be documented in a detailed report to the NRC, as required by 10 CFR 72.54 (Ref. 4).

Several different surveys are typically required as part of the decommissioning process. Because each is intended to provide radiological data for different primary applications or objectives, the survey techniques, thoroughness, data accuracy, and documentation requirements may vary. Because of the importance of the final survey, the final survey plan will include at least the following:

- types, numbers, and locations of measurements and samples to be obtained
- equipment, calibration and testing, and techniques that will be used for measuring, sampling, and analyzing data
- interpretation and evaluation methodologies for the data
- quality assurance methods for ensuring the validity of the data

Representative samples will be collected from below-grade areas before the areas are filled; the sample locations will either be randomly selected or will be selected based on facility operating records. Allowable radioactivity levels for below-grade areas will be based on modeling using approved computer dose models and may be different than allowable above-grade levels. However, the same release criteria will be used throughout the facility.

The final survey plan will be based on guidance provided in NUREG/CR-5849 (Ref. 6), including guidance on quality control/assurance, selection of measurement/sampling locations, and sampling frequency to ensure the statistical significance of the data.

5.0 DECOMMISSIONING COST ESTIMATE AND FUNDING PLAN

In accordance with 10 CFR 72.30(b) (Ref. 4), this section provides:

- an estimate of the ISF Facility decommissioning costs
- a funding plan

5.1 DECOMMISSIONING COST ESTIMATE

This section provides a cost estimate prepared by FWENC for the decommissioning of the ISF Facility and describes the basis for the estimate. The costs of activities involved in radiological decommissioning as well as expenditures necessary to complete non-radiological site restoration activities are included in the cost estimate. The costs of removal and disposal of non-radioactive structures and materials are identified separately from the costs of radiological decommissioning.

5.1.1 Cost Estimate

The costs (in 2001 dollars) for the selected decommissioning alternative have been estimated at approximately $22,600,000 for radiological decommissioning activities and approximately $13,200,000 for non-radiological decommissioning activities (site restoration).

5.1.2 Cost Estimate Description

The methodology used to develop the cost estimate followed the approach presented in AIF/NESP-036 (Ref. 10) and the DOE's *Decommissioning Handbook* (Ref. 11). These guidance documents use a unit-cost factor method for estimating decommissioning activity costs. Unit cost factors incorporate site-specific considerations whenever practicable. Quantities and volumes of the equipment and material expected to be removed during decommissioning were estimated using proposed facility drawings. Unit cost factors were applied to the quantities to estimate the "activity-dependent" costs. "Period-dependent" costs were determined from a critical path schedule based on the duration of removal activity. The cost estimate also includes appropriate "peripheral" costs (e.g., work plans, procedures, engineering) and "waste" costs as described in the DOE *Decommissioning Handbook* (Ref. 11).

5.1.2.3 Radiological Decommissioning Costs

The radiological decommissioning cost estimate for the ISF Facility is provided in the following table. Consistent with current NRC policy, the radiological decommissioning cost estimate for the ISF Facility considers radiological decommissioning costs to be only those costs associated with normal decommissioning activities necessary for the release of the site for unrestricted use. The radiological decommissioning cost estimate does not include those costs associated with spent nuclear fuel management or the disposal of non-radioactive structures and materials.

Burial costs were derived from FWENC modeling and analysis of low-level radioactive waste disposal costs. Contingencies were applied to each area of the cost estimate (i.e., decontamination and dismantlement, waste disposal, final survey). No credit was taken for equipment salvage value.

Estimate of Decommissioning Costs (2001 dollars)

Radiological (NRC) Decommissioning Costs	
Dismantlement, Decontamination, and Remediation	12,500,000
Waste Disposal	6,300,000
Final Survey	3,800,000
Subtotal	22,600,000
Non-Radiological Decommissioning Costs	
Site Restoration Total	13,200,000
Total Decommissioning Cost	**35,800,000**

NOTES: Waste disposal estimate based on:
- Construction debris landfill at $16 to $22 per ton
- Low-level radioactive waste at $1360 to $5000 per ton
- Special materials at $ 37,500 per ton

5.1.2.4 Non-Radiological Decommissioning Costs

Although not required by NRC regulations, the decommissioning cost estimate includes non-radiological decommissioning costs. The cost estimate considers non-radiological decommissioning costs to be those costs associated with site remediation and demolition and removal of uncontaminated structures.

5.2 DECOMMISSIONING FUNDING PLAN

Regulation 10 CFR 72.30(c) (Ref. 4) provides financial assurance methods acceptable for decommissioning. The ISF Facility is designed, licensed, constructed, and operated under a privatization contract between FWENC and the DOE (Ref. 1). Decommissioning of the facility will remain the responsibility of the DOE in accordance with Section H.19, Decontamination and Decommissioning, of the contract, which contains the following statement:

> *The Department of Energy will maintain responsibility for future facility decontamination and decommissioning when they are no longer useful. The Government, however, may exercise the option to require the Contractor to decontaminate and decommission the facilities...*

Regulation 10 CFR 72.30(c)(4) (Ref. 4) provides a financial assurance method available to government agencies. This assurance method allows government agencies to provide a statement of intent to obtain required funding when necessary.

Section B.9, Funding Obligations, of the contract between FWENC and the DOE states:

> *...The parties contemplate that the Department of Energy will obligate additional funds incrementally to the contract as necessary to secure timely contract performance in accordance with the contract schedule and to meet the Government's termination liability requirements in the event of a termination for convenience. The Contractor shall provide an annual update of funding profile and schedule of values, required elsewhere in this contract, by March 1 of each year, which will be used as the basis for the Department of Energy's budget and appropriation requests to Congress. The Department of Energy*

shall endeavor to give this contract a high priority within its appropriated funds for each year of contract performance; provided, however, that nothing in this contract shall be considered to bind or otherwise obligate the Congress to appropriate funds sufficient to cover the contract requirements.

Section B.9 of the DOE contract with FWENC provides the statement of intent by the DOE required by 10 CFR 72.30(c)(4) (Ref. 4).

The DOE maintains a current copy of this contract on the DOE Idaho Operations Office website. The contract is available for viewing at http://www.id.doe.gov/doeid/psd/SNFDSPContract.htm.

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6.0 PHYSICAL PROTECTION PLAN PROVISIONS

The Physical Protection Plan is based on the requirements of 10 CFR 73.51 (Ref. 12) and NRC-approved exemptions thereto. The Physical Protection Plan describes the overall organization and protective systems used to protect the ISF Facility Site against unauthorized access, sabotage, and/or other potential security contingencies associated with the storage of spent nuclear fuel.

Once the spent nuclear fuel is removed from the ISF Facility Site, the specific purpose and mode of operation of the ISF Facility will change considerably. A scheme for reducing accountability and control for nuclear material, as well as safeguards associated with both nuclear material and sensitive equipment will need to be included in the final decommissioning planning and evaluated in accordance with 10 CFR 72.48 (Ref. 4).

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7.0 REFERENCES

1. DOE-ID (2000), *Contract Award and Notice to Proceed*, Contract No. DE-AC07-001D13729, Spent Nuclear Fuel Dry Storage Project, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho, May.

2. DOE, *Conceptual Plan for Decommissioning the INEL TMI-2 Independent Spent Fuel Storage Installation*, Rev. 0, Submitted as Part of TMI-2 ISFSI License Application, Docket No. 72-20, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho.

3. U.S. Nuclear Regulatory Commission (2000), NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, Washington, D.C.

4. Title 10, Code of Federal Regulations, Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, Washington, D.C.

5. U.S. Nuclear Regulatory Commission (1988), NUREG-0586, Final Generic Environmental Impact Statement on Decommissioning Nuclear Facilities, Washington, D.C.

6. U.S. Nuclear Regulatory Commission (1992), NUREG/CR-5849, *Manual for Conducting Radiological Surveys in Support of License Termination*, Washington, D.C.

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8. U.S. Nuclear Regulatory Commission (1994), NUREG-1500, *Working Draft of Release Criteria for Decommissioning*, Washington, D.C.

9. U.S. Nuclear Regulatory Commission (1974), Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors*, Washington, D.C.

10. TLG Engineering (1986), Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates, AIF/NESP-036.

11. DOE (1994), *Decommissioning Handbook*, DOE/EM-0142P, U.S. Department of Energy, Washington, D.C.

12. Title 10, Code of Federal Regulations, Part 73, *Physical Protection of Plants and Materials,* Washington, D.C.

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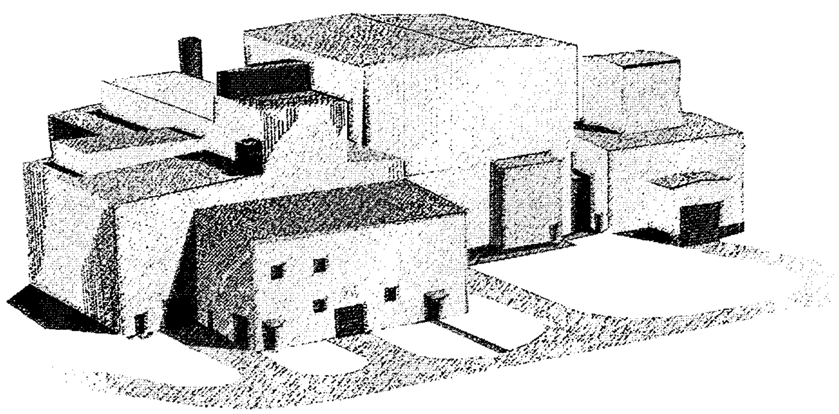
License Application

Appendix D

Proposed Technical Specifications

Idaho Spent Fuel Facility

Docket No. 72-25



ISF-FW-RPT-0034

FOSTER WHEELER ENVIRONMENTAL CORPORATION

PROPOSED TECHNICAL SPECIFICATIONS

FOR

ISF FACILITY

Technical Specifications
Table of Contents

1.0 USE AND APPLICATION

1.1 Definitions

-- NOTE --

The defined terms of this section appear in capitalized type and are applicable throughout these ISF Facility Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under the designated Conditions within the specified Completion Times.
CANISTER HANDLING	CANISTER HANDLING exist when SPENT NUCLEAR FUEL is contained in an ISF CANISTER that has passed its leak rate acceptance test and is not within a STORAGE TUBE that has passed its interseal leak rate acceptance test.
CHANNEL CHECK	A CHANNEL CHECK is the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications and status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	A CHANNEL FUNCTIONAL TEST is the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display, and trip functions.
ISF CANISTER	The ISF CANISTER is the sealed SPENT NUCLEAR FUEL container that consists of a cylindrical shell with welded upper and lower closure heads. The ISF CANISTER provides for the canning of consolidated fuel rods or unconsolidated assemblies to meet the requirements of 10 CFR 72.122(h)(1). The ISF CANISTER also provides the primary confinement for the stored SPENT NUCLEAR FUEL.

(continued)

1.1 Definitions (continued)

LOADING OPERATIONS	LOADING OPERATIONS include activities associated with packaging SPENT NUCLEAR FUEL into ISF canisters. LOADING OPERATIONS exist whenever • SPENT NUCLEAR FUEL is present in a transfer cask without a fully tensioned closure lid; • SPENT NUCLEAR FUEL is in the Fuel Packaging Area; or • SPENT NUCLEAR FUEL is in an ISF CANISTER that has not completed its leak rate acceptance test.
OPERABLE/OPERABILITY	A system, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, and other auxiliary equipment required for the system, component, or device to perform its specified important to safety function(s) are also capable of performing their related support functions.
RECEIPT OPERATIONS	RECEIPT OPERATIONS include all activities associated with handling SPENT NUCLEAR FUEL while it is contained in a transfer cask. Receipt operations begin when the transfer cask is received at the ISF Facility and end when the first bolt on the transfer cask lid is loosened.
SPENT NUCLEAR FUEL	SPENT NUCLEAR FUEL means fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor and has not been chemically separated into its constituents elements by reprocessing. SPENT NUCLEAR FUEL includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies.
STORAGE OPERATIONS	STORAGE OPERATIONS exist when an ISF CANISTER is contained within a STORAGE TUBE that has passed its interseal leak rate acceptance test.
STORAGE TUBE	The STORAGE TUBE is the sealed ISF CANISTER container, which consists of a cylindrical shell, shield plug, and a bolted closure plate. The STORAGE TUBE provides the secondary confinement boundary for the stored radioactive materials.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that may appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Completion Time, Surveillance, or Frequency.

1.2 Logical Connectors (continued)

EXAMPLES The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met	A.1 Verify.... AND A.2 Restore...	

In this example the logical connector AND is used to indicate that when in Condition A, both required Actions A.1, and A.2 must be completed.

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met	A.1 Stop.... OR A.2.1 Verify... AND A.2.2.1 Reduce... OR A.2.2.2 Perform... OR A.3 Remove...	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operations (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure with Completion Times based on initial entry into the Condition.

(continued)

1.3 Completion Times (continued)

EXAMPLES The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions..

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Times not met	B.1 Perform Action B.1 AND	12 hours
	B.2 Perform Action B.2	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limits.	A.1 Restore system to within limit	7 days
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1 AND	12 hours
	B.2 Complete action B.2	36 hours

When a system is determined to not meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

1.3 Completion Times (continued)

EXAMPLE 1.3-3

-- NOTE --
Separate Condition entry is allowed for each component.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Complete action B.1 AND B.2 Complete action B.2	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only required when it can be and should be performed. With a SR satisfied, SR 3.0.4 imposes no restriction.

(continued)

1.4 Frequency (continued)

EXAMPLES The following examples illustrate the various ways that Frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify pressure within limit.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed within 12 hours prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

2.0 APPROVED CONTENTS

2.1 The ISF Facility shall be limited to the receipt, packaging, and storage of the following SPENT NUCLEAR FUEL:

- Peach Bottom fuel elements with characteristics as described in Table 2-1,
- Shippingport fuel rods with characteristics as described in Table 2-2, and
- TRIGA fuel elements with characteristics as described in Table 2-3.

Table 2-1.
Spent Fuel Limits – Peach Bottom Fuel

Characteristic	Limit
Cladding	Graphite
Maximum Fuel Enrichment	16 w/o uranium enriched to 93.15% ^{235}U
Maximum Decay Heat per ISF CANISTER	33 W
Fuel Design	High-temperature gas-cooled with graphite moderation
Maximum Burnup	900 EFPD

Table 2-2.
Spent Fuel Limits –Shippingport LWBR Fuel

Characteristic	Limit
Cladding	Zircaloy-4
Maximum Fuel Enrichment	Not applicable
Maximum Decay Heat per ISF CANISTER	10 W
Fuel Design	LWBR ThO_2 Type IV/V reflector
Maximum Burnup	30,000 EFPH

(continued)

2.0 APPROVED CONTENTS (continued)

Table 2-3.
Spent Fuel Limits – Training Research Isotope Production General Atomics (TRIGA) Fuel

Characteristic	Limit
Cladding	Aluminum or Stainless Steel
Maximum Enrichment	9 w/o uranium enriched to 20% ^{235}U
Maximum Decay Heat per ISF CANISTER	36 W

2.2 The decay heat load of the ISF Facility storage vaults shall not exceed the limits shown in Table 2-4.

Table 2-4
ISF Facility
Heat Load Limits

	STORAGE TUBE Heat Load (Watts)	Number of STORAGE TUBES	Heat Load (Watts)	Vault Heat Load (Watts)
Vault 1	40	76	3040	6160
	120	26	3120	
Vault 2	40	132	5280	6720
	120	12	1440	
Total				12880

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Actions(s) is not required, unless otherwise stated.
LCO 3.0.3	Not applicable to an ISFSI.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS.
LCO 3.0.5	Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a SR, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per ..." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's SRs have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS or that are related to unloading of a STORAGE TUBE.

3.1 Canister Integrity

3.1.1 Canister Integrity

LCO 3.1.1 The ISF CANISTER helium leak rate shall be $\leq 10^{-4}$ std cm^3/sec at a pressure of 19-21 psia at 80-100°F.

APPLICABILITY: CANISTER HANDLING, STORAGE OPERATIONS

ACTIONS:

---NOTE---
Separate Condition entry is allowed for each ISF Canister.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Establish ISF CANISTER pressure and fill helium leak rate within limits.	Prior to CANISTER HANDLING

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.	Verify ISF CANISTER fill pressure and helium leak rate is within limits.	Once prior to CANISTER HANDLING.

3.2 STORAGE TUBE Integrity

3.2.1 STORAGE TUBE Pressure and Interseal Leak Rate

LCO 3.2.1 The STORAGE TUBE interseal leak rate shall be $\leq 10^{-4}$ std cm^3/sec at a pressure within the limits of Figure 3.2-1.

APPLICABILITY: STORAGE OPERATIONS

ACTIONS:

--NOTE--

Separate Condition entry is allowed for each STORAGE TUBE containing an ISF CANISTER.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore STORAGE TUBE pressure and interseal leak rate within limits.	30 days

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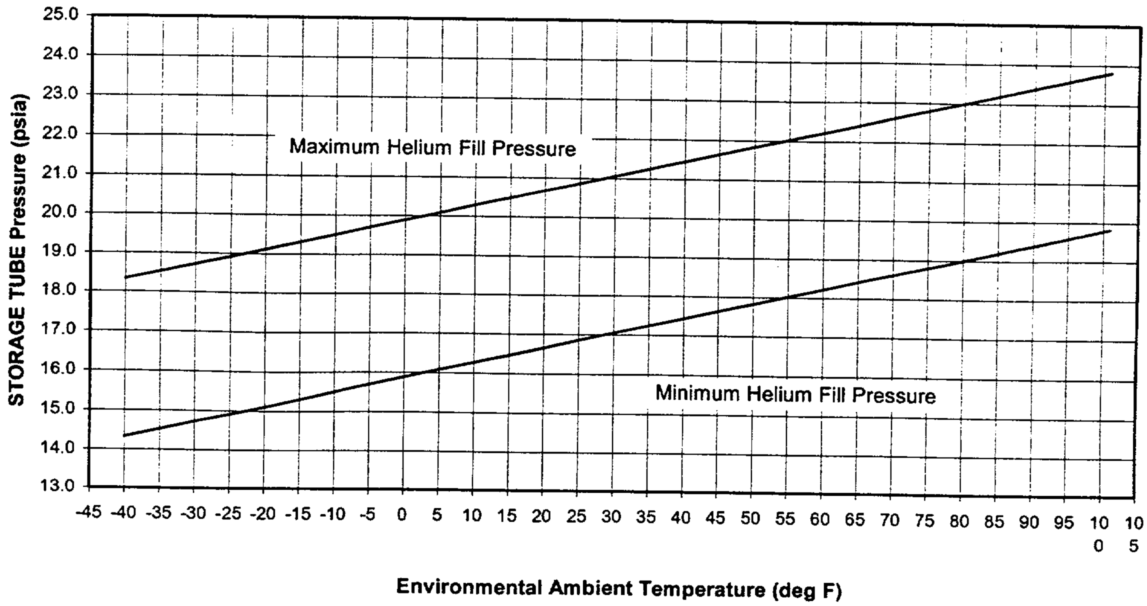
3.2 STORAGE TUBE Integrity

Figure 3.2-1
STORAGE TUBE Helium Fill Pressure Limits



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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify STORAGE TUBE pressure and interseal leak rate within specified limits.	Prior to commencing STORAGE OPERATIONS for the STORAGE TUBE being tested.
SR 3.2.1.2 Verify STORAGE TUBE pressure and interseal leak rate within specified limits.	Annually for 1 occupied STORAGE TUBE from each storage vault.

3.2 STORAGE TUBE Integrity

3.2.2 Storage Vault Heat Removal System

LCO 3.2.2 The Storage Vault Heat Removal System for each occupied STORAGE TUBE shall be OPERABLE.

APPLICABILITY: STORAGE OPERATIONS

ACTIONS:

--NOTE--
Separate Condition entry is allowed for each STORAGE TUBE containing an ISF CANISTER.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. LCO not met.	A.1	Restore Storage Vault Heat Removal System to OPERABLE status.	48 hours
B. Required Action and associated Completion Time not met.	B.1	Transfer the ISF CANISTER to an unaffected STORAGE TUBE.	96 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.	Visually inspect all inlet air vents, outlet air vents, and occupied STORAGE TUBE annular outlets for blockage.	48 hours

3.3 Criticality Control Program

3.3.1 Fuel Packaging Area Limits

LCO 3.3.1 Only one SPENT NUCLEAR FUEL type (i.e., Peach Bottom, Shippingport, or TRIGA) shall be present within the Fuel Packaging Area.

APPLICABILITY: RECEIPT OPERATIONS, LOADING OPERATIONS

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Suspend LOADING OPERATIONS.	Immediately
	AND	
	A.2 Perform an evaluation and develop a recovery plan to restore Fuel Packaging Area such that it contains only one SPENT NUCLEAR FUEL type.	72 hours
	AND	
	A.3 Restore Fuel Packaging Area such that only one SPENT NUCLEAR FUEL type is present	90 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Verify SPENT NUCLEAR FUEL type in transfer cask.	Prior to transferring transfer cask loaded with SPENT NUCLEAR FUEL into Transfer Tunnel
SR 3.3.1.2	Verify SPENT NUCLEAR FUEL type in Fuel Packaging Area	Once during visual inspection of first fuel package unloaded from each transfer cask.

3.3 Criticality Control Program

3.3.2 Criticality Monitoring

LCO 3.3.2 The criticality monitoring system shall be OPERABLE.

APPLICABILITY: LOADING OPERATIONS with SPENT NUCLEAR Fuel in the Fuel Packaging Area.

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Suspend LOADING OPERATIONS. AND A.2 Restore criticality monitoring system to operation.	Immediately Prior to resuming fuel movement.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform a CHANNEL CHECK of criticality monitoring system.	Within 1 hour prior to commencing LOADING OPERATIONS in the Fuel Packaging Area and every 24 hours thereafter.
SR 3.3.2.2	Perform a CHANNEL FUNCTIONAL TEST of criticality monitoring system.	Once, within 7 days prior to initial LOADING OPERATIONS in the Fuel Packaging Area and every 12 months thereafter.

3.4 Fuel Packaging Area Confinement Boundary

3.4.1 Heating, Ventilation, Air Conditioning (HVAC) System

LCO 3.4.1 HVAC System shall be OPERABLE.

APPLICABILITY: LOADING OPERATIONS

ACTIONS:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Suspend SPENT NUCLEAR FUEL movement.	Immediately
	AND	
	A.2 Verify all supply fans deenergized	Immediately
	AND	
	A.3 Ensure confinement penetration boundaries closed.	1 hour
	AND	
	A.3 Commence air monitoring.	Within 1 hour of HVAC system declared INOPERABLE and every 8 hours thereafter.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR.3.4.1.1	Verify 1 Fuel Packaging Area exhaust fan running.	Within 1 hour prior to commencing LOADING OPERATIONS. <u>AND</u> Every 24 hours during LOADING OPERATIONS.
SR 3.4.1.2	Verify HVAC primary exhaust HEPA filter differential pressure is >0 in w.g. and < 4 in w.g.	Within 1 hour prior to commencing LOAD OPERATIONS. <u>AND</u> Every 24 hours during LOADING OPERATIONS.
SR 3.4.1.3	Verify the following Fuel Packaging Area access port conditions: Cask port plug in place <u>OR</u> Transfer cask positioned beneath cask port and associated seal inflated <u>AND</u> Canister port plug in place <u>OR</u> Canister cask positioned beneath canister port and associated seal inflated. <u>AND</u> Waste port plugs in place <u>OR</u> SPENT NUCLEAR FUEL in designated storage locations in Fuel Packaging Area.	Within 1 hour prior to commencing LOADING OPERATIONS with SPENT NUCLEAR FUEL in the Fuel Packaging Area. <u>AND</u> Every 24 hours when SPENT NUCLEAR FUEL is in the Fuel Packaging Area.

4.0 DESIGN FEATURES

4.1 Design Features Significant to Safety

4.1.1 Criticality Control

ISF CANISTER loading shall not exceed the following limits:

SPENT NUCLEAR FUEL	Maximum Loading per ISF CANISTER
Peach Bottom	10 elements
TRIGA	108 elements
Shippingport LWBR	1 reflector module or 127 loose rods

4.1.2 Materials

1. Confinement boundary materials

   The STORAGE TUBE and lid shall be constructed of carbon steel to form a pressure vessel.

   The ISF CANISTER and lid shall be constructed of stainless steel to form a pressure vessel.

2. Confinement boundary seals

   During STORAGE OPERATIONS, SPENT NUCLEAR FUEL shall be confined in a welded ISF CANISTER within a bolted STORAGE TUBE employing redundant ring seals.

4.0 DESIGN FEATURES

4.2 Codes and Standards

The following are the governing codes for the ISF Facility storage component design:

Storage Component Important to Safety	Applicable Codes	Editions / Years	Application	
			Design	Fabrication
STORAGE TUBE	ASME Boiler and Pressure Vessel Code (B&PVC), Section II ASME B&PVC, Section III, Division 1, Subsection NCA, NC, and Appendix F ASME B&PVC, Section V ASME B&PVC, Section IX	1998 with 2000 addenda	Yes	Yes
ISF CANISTER	ASME B&PVC, Section II ASME B&PVC, Section III, Division 1, Subsections NCA, NB, Appendix F ASME B&PVC, Section V ASME B&PVC, Section IX	1998 with 2000 addenda	Yes	Yes
ISF Basket	ASME B&PVC, Section II, ASME B&PVC, Section III, Division 1, Subsections NCA, NF, NG, & Appendix F	1998 with 2000 addenda	Yes	No. In accordance with ISF Quality Program Plan

4.2.1 Alternatives to Codes, Standards, and Criteria

No alternatives to the codes listed in 4.2 above have been used in the design of the ISF Facility

4.2.2 Construction/Fabrication Alternatives to Codes, Standards, and Criteria

Proposed alternatives to the codes listed in 4.2 above may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternatives should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or
2. Compliance with the specified requirements of the codes listed in 4.2 above would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for alternatives shall be submitted in accordance with 10 CFR 72.4

4.0 DESIGN FEATURES

4.3 SPENT NUCLEAR FUEL Handling Equipment

4.3.1 ISF Facility Cranes and Trolleys

The components classified to be important to safety of the Cask Receipt Crane, the Fuel Handling Machine, the Canister Handling Machine, the Cask Trolley, and the Canister Trolley shall meet the requirements of NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants", and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants".

4.3.2 Lifting Devices

All lifting devices used to raise or lower SPENT NUCLEAR FUEL outside of the FPA shall be designed in accordance with ANSI N14.6 – 1993, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More".

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The ISF Facility Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The ISF Facility Manager or his designee shall approve, prior to implementation, each proposed change, test, or experiment to structures, systems, or components that are important to safety as defined in 10 CFR 72.3.

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for facility operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the ISF Facility.

a) Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and operating organization positions. These relationships shall be documented and updated, as appropriate, in organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the Safety Analysis Report;

b) The ISF Facility Manager shall be responsible for overall safe operation of the facility and shall have control over those onsite activities necessary for safe operation and maintenance of the facility;

c) A designated corporate executive shall have corporate responsibility for overall facility nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the facility to ensure nuclear safety; and

d) The individuals who perform health physics functions, or perform quality assurance functions may report to the ISF Facility Manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.0 ADMINISTRATIVE CONTROLS

5.3 ISF Facility Staff Qualifications

5.3.1 The ISF Facility Staff shall meet or exceed the minimum qualifications of ANSI 18.1-1971 for comparable positions. The ISF Facility Operations Manager and certified Operators shall be trained and certified in accordance with the ISF Operator Training Plan.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities that are important to safety:

a) Administrative controls;

b) Routine ISF Facility operations;

c) Alarms and Annunciators;

d) Emergency operations;

e) Design control and facility change or modification;

f) Control of surveillances and tests;

g) Control of special processes;

h) Maintenance;

i) Health physics, including ALARA practices;

j) Special nuclear material accountability;

k) Quality assurance, inspection, and audits;

l) Physical security and safeguards;

m) Records management;

n) Reporting; and

o) All programs specified in Specification 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs

The following programs shall be established, implemented, and maintained.

5.5.1 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a) Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b) Licensees may make changes to the Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the SAR or Bases that involves an unreviewed safety question, a significant increase in occupational exposure, or a significant unreviewed environmental impact as defined in 10 CFR 72.48.

c) The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the SAR.

d) Proposed changes that do not meet the criteria of 5.5.1.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 72.48 (b) (2).

5.0 ADMINISTRATIVE CONTROLS

5.5.2 Radioactive Effluent Control Program

This program contains the offsite dose calculation methodologies, radioactive effluent controls programs, and radiological monitoring activities. This program shall contain:

a) The methodologies and parameters used in calculation of offsite doses resulting from radioactive gaseous and liquid effluents;

b) The methodologies and parameters used in calculation of gaseous and liquid effluent monitoring alarm and trip setpoints;

c) The controls for maintaining the doses to members of the public from radioactive effluents as low as is reasonably achievable in accordance with 10 CFR 72.104(b). These include:

1. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance test and setpoint determination;

2. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302

Changes to the program shall be documented and records of reviews performed shall be retained. This documentation shall contain:

a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

b) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, and 10 CFR 72.104.

5.0 ADMINISTRATIVE CONTROLS

5.5.3 Fuel Handling Program

This program implements the ISF Safety Analysis Report requirements for receipt, packaging, and storage of SPENT NUCLEAR FUEL. At a minimum, the program shall establish criteria that need to be verified to address ISF Facility Safety Analysis commitments and regulatory requirements for:

a) Transfer Cask and fuel acceptance criteria;

b) Fuel Packaging Area limits to ensure restrictions on fuel types are not violated;

c) Limiting operations of structures, systems, or components that are important to safety to certified operators qualified in accordance with the ISF Facility Operator Training Plan.

d) Helium inerting pressure and purity to assure corrosion control;

e) Leak testing to assure adequate ISF CANISTER and STORAGE TUBE integrity and consistency with the offsite dose analysis; and

f) Configuring the Fuel Packaging Area for the fuel type being packaged.

The program shall include compensatory measures and appropriate completion times if program requirements are not met.

5.0 ADMINISTRATIVE CONTROLS

5.5.4 Fire Protection Program

This program contains the fire protection policy for protection of structures, systems, and components important to safety at the ISF Facility and the procedures, equipment, and personnel required to implement the program at the facility. At a minimum, the program shall contain:

a) Organizational structure for fire protection responsibilities including design, maintenance, surveillance, quality assurance of fire protection features, fire prevention activities, and fire fighting organization and training.

b) Fire Hazards Analysis describing the defense-in-depth approach for fire areas important to safety.

c) Implementing procedures for surveillance of fire protection equipment including identification of suitable compensatory measures for degraded or inoperable components.

d) Fire pre-plans for fire fighting strategies.

e) Administrative controls for housekeeping, control of combustibles, control of ignition sources (hot work), and fire notification.

5.0 ADMINISTRATIVE CONTROLS

5.5.5 Radiation Protection Program

This program contains the radiation protection policy for maintaining onsite and offsite personnel exposure as low as is reasonably achievable (ALARA). At a minimum, the program shall contain:

a) Procedures and administrative controls to limit personnel exposure ALARA in accordance with 10 CFR 20.

b) Requirements for monitoring the DOE transfer cask during RECEIPT and LOADING OPERATIONS to ensure that surface dose rates are within analyzed values.

c) A monitoring program to ensure the annual dose equivalent to any real individual located outside the ISF Facility controlled area does not exceed regulatory limits is incorporated as part of the environmental monitoring program in the Radioactive Effluent Control Program of Specification 5.5.2.

d) Requirements for monitoring the DOE transfer cask during RECEIPT and LOADING OPERATIONSprior to and after unloading SPENT NUCLEAR FUEL to ensure that removable surface contamination levels do not exceed 2200 dpm/100 cm^2 from beta and gamma sources and 220 dpm/100 cm^2 from alpha sources.

e) Measures for controlling access to high radiation areas as defined by 10 CFR 20. These measures are alternative methods allowed by 10 CFR 20.1601(c) and further described in Regulatory Position 2.4 of Regulatory Guide 8.38, *Control of Access to High and Very High Radiation Areas in Nuclear Power Plants.*

   Each area, accessible to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent in excess of 0.1 rem (100 mrem) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto should be controlled by requiring issuance of a radiation work permit (RWP) or equivalent.

   Radiation Control Technicians or other individuals trained and qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempted from this RWP requirement while performing their assigned duties in high radiation areas where radiation doses could be received that are equal to or less 1.0 rem in 1 hour (measured at 30 centimeters from any source of radiation) provided they are otherwise following plant radiation protection procedures, or a general radiation protection RWP, for entry into such high radiation areas.

   Any individual or group of individuals permitted to enter such areas should be provided with or accompanied by one or more of the following:

- A radiation monitoring device that continuously indicates the radiation dose rate in the area,

- A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rates in the area have been determined and personnel have been made knowledgeable of them,

- An individual qualified in radiation protection procedures with a radiation dose rate monitoring device. This individual is responsible for providing positive radiation protection control over the activities within the area and should perform periodic radiation surveillance at the frequency specified in the radiation protection procedures or the applicable RWP.

In addition, areas that are accessible to personnel and that have radiation levels greater than 1.0 rem (but less than 500 rads at 1 meter) in 1 hour at 30 cm from the radiation source, or from any surface penetrated by the radiation, should be provided with locked doors to prevent unauthorized entry, and the keys should be maintained under the administrative control of the shift supervisor on duty or health physics supervisor. Doors should remain locked except during periods of access by personnel under an approved RWP that specifies the dose rates in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of a stay time specification on the RWP, direct or remote continuous surveillance (such as closed circuit TV cameras) may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

Individual high radiation areas that are accessible to personnel, that could result in radiation doses greater than 0.01 Sv (1.0 rem) in 1 hour, and that are within large areas where no enclosure exists to enable locking and where no enclosure can be reasonably constructed around the individual area should be barricaded and conspicuously posted. A flashing light should be activated as a warning device whenever the dose rate in such an area exceeds or is expected to exceed 1.0 rem in 1 hour at 30 cm from the radiation source or from any surface penetrated by the radiation.

The Radiation Protection Program will be reviewed annually for content and implementation.

IFS Facility
Technical Specification Bases

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to operable status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a cessation of operations may be required to place the system or component in a condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Time of the Required Actions are also applicable when a system or component is removed from service intentionally. The reason for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

LCO 3.0.3

This specification is not applicable to an ISFSI. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the facility being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with the Required Actions that permit continued operation of the facility for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the facility. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to establishing and maintaining the spent fuel in an inert atmosphere.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. These exceptions allow entry into specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The equipment being returned to service meets the LCO: or

b. Other equipment meets the applicable LCOs.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed surveillance. This Specification does not provide time to perform any other preventive or corrective maintenance.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify systems, components, and variables are within specified limits. Failure to meet a SR within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances. Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to service.

Upon completion of maintenance, appropriate post maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary facility parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of changes in the specified conditions in the Applicability imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as a convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Time of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Time of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe operation of the facility.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem, component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on such equipment. When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillances(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs annotation is found in Section 1.4, Frequency.

3.1 CANISTER/ INTEGRITY

3.1.1 Canister Integrity

BACKGROUND

After receiving SPENT NUCLEAR FUEL from the Fuel Packaging Area, the ISF CANISTER is moved through the Transfer Tunnel to the Canister Closure Area (CCA). The canister is enclosed by a welded lid, vacuum dried, and backfilled with helium to 19-21 psia at 80-100°F. During storage, the inert helium gas inhibits corrosion within the canister and assists heat transfer from the spent fuel. The integrity of the canister lid welds is tested by a helium leak detector to ensure that the canister leak rate is $\leq 10^{-4}$ std cm^3/sec.

APPLICABLE SAFETY ANALYSIS

The ISF CANISTER is the primary confinement barrier against the migration of radiation during the storage of SPENT NUCLEAR FUEL. An inert, non-corrosive atmosphere promotes long-term integrity of the canister and the fuel. For this reason, the canister is backfilled with helium prior to final sealing in the CCA. Helium also promotes the transfer of decay heat from the fuel to the canister wall. Thermal analysis of the ISF CANISTER demonstrates that helium is not essential for safe canister handling and storage operations.

During canister closure in the CCA thermal analysis of the fuel assumes the fuel elements to be surrounded by air. The results of this analysis show that, for the off-normal condition of Transfer Tunnel temperature at 163°F, the maximum temperature attained by TRIGA elements, the most limiting fuel type, is approximately 187°F. This temperature is well below the normal temperature limit (400°F) for TRIGA fuel and produces temperatures in the surrounding fuel basket, canister, cask, and concrete that are within their design limits. Therefore, spent fuel in an air environment within the canister cask can remain in the CCA indefinitely without exceeding fuel or supporting component temperature limits.

LCO

Verifying the ISF CANISTER leak rate to be no greater than 10^{-4} std cm^3/sec satisfies ASME code requirements for acceptable leakage. The specified canister pressure of 19-21 psia ensures that the canister internal pressure will always be greater than barometric atmospheric pressure under all expected storage temperatures.

APPLICABILITY

The ISF CANISTER receives its helium backfill near the end of LOADING OPERATIONS. Therefore, the need for the canister to maintain the backfill applies only to subsequent activities, namely CANISTER HANDLING and STORAGE OPERATIONS.

ACTIONS

A Note added to the ACTIONS states that a separate Condition entry is allowed for each ISF CANISTER. Subsequent ISF CANISTERS that do not meet the LCO are governed by subsequent Condition entries and application of the associated Required Actions.

A.1

If the ISF CANISTER leak rate exceeds 10^{-4} std cm^3/sec at 19-21 psia, action must be taken to return the fill pressure and leak rate to within that limit prior to placing the canister into storage. The ACTION statement does not require a specific completion time but only that the canister leak rate be restored to within the limits prior to CANISTER HANDLING.

Typically, the helium backfill is used to promote heat transfer from the SPENT NUCLEAR FUEL to the canister wall. Thermal analysis, however, shows that all three types of fuel can remain surrounded by air or helium indefinitely in the ISF CANISTER and the canister cask without exceeding temperature limits. Therefore, temperature considerations do not impose any time limit for reestablishing backfill pressure or canister integrity.

The helium backfill also provides an inert atmosphere to impede corrosion within the ISF CANISTER. Because the stainless steel construction of the canister is highly resistant to corrosion and the ambient environment of the Idaho desert is typically dry, the formation of corrosion within the ISF CANISTER is unlikely during the short period of time needed to repair or replace it. Therefore, corrosion considerations place no time limit on the completion time of the ACTION provided that the leak rate is restored to within its limit prior to CANISTER HANDLING.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

During LOADING OPERATIONS, the canister lid is welded to the ISF CANISTER and the canister is evacuated to remove residual moisture, purged with helium, evacuated and purged a second time. During the helium purge, the canister is pressurized to 19-21 psia while the lid weld is inspected for leaks. The vent plug is then torqued, welded, and given a final leak check. Any leak larger than 10^{-4} std cm^3/sec requires the canister to be repaired or replaced.

The correct canister pressure and leak rate is verified prior to STORAGE OPERATIONS to ensure that the ISF CANISTER contains an inert atmosphere prior to placement in a STORAGE TUBE. Once a canister is stored in a pressurized STORAGE TUBE, verifying its internal pressure is impractical.

REFERENCES

ASME Boiler and Pressure Vessel Code (1998), Section V
ISF Facility Safety Analysis Report, Chapter 8

3.2 STORAGE TUBE INTEGRITY

3.2.1 STORAGE TUBE Pressure and Interseal Leak Rate

BACKGROUND

After being pressurized with helium and sealed, the ISF CANISTER is moved by the canister trolley to the Storage Area where the Canister Handling Machine removes it from the trolley and lowers it into a STORAGE TUBE. In preparation for STORAGE OPERATIONS, the tube is evacuated, backfilled with helium, and sealed to provide a dry, inert atmosphere that prevents degradation of the canister.

APPLICABLE SAFETY ANALYSIS

The helium backfill of the STORAGE TUBE serves to:

1. provide a secondary confinement atmosphere that is consistent with the primary confinement atmosphere; and
2. provide an inert atmosphere within the tube to prevent corrosion.

The STORAGE TUBE provides the secondary confinement barrier for the SPENT NUCLEAR FUEL and the internal helium atmosphere of the tube protects the stored ISF CANISTER from unacceptable degradation. To achieve this, the helium pressure within the tube must be maintained above atmospheric pressure to prevent the incursion of air. Both the stainless steel canister and the coated carbon steel interior of the tube are resistant to corrosion. Therefore, short-term exposure of these surfaces to air, such as during loading and backfilling operations, is permissible.

The enhanced heat transfer ability between the canister and the STORAGE TUBE wall afforded by the helium backfill is a relatively minor effect. Thermal analysis has shown that, under accident conditions, the heatup of TRIGA fuel, the most limiting fuel type, results in maximum fuel and structural temperatures that are well within design limits. This analysis conservatively took no credit for helium or air within the STORAGE TUBE as a heat removal medium. Consequently, the presence of helium is useful but not necessary in promoting heat transfer between the ISF CANISTER and the STORAGE TUBE.

Figure 3.2-1 of the Technical Specifications provides the maximum and minimum allowable helium fill pressures for a STORAGE TUBE. The minimum pressure is based upon establishing an internal tube pressure that remains above atmospheric pressure for the anticipated temperature range of outside air. Maintaining tube pressure above barometric atmospheric pressure confirms that helium is present and providing an inert atmosphere to prevent corrosion.

The maximum pressure limit of Figure 3.2-1 is based upon preventing an excessive differential pressure across the walls of the ISF CANISTER within the STORAGE TUBE. Restricting the pressure of the tube helium backfill to maximum shown on the figure ensures that the canister will not be subjected to excessive compressive forces during any anticipated storage condition.

LCO

Verifying the ISF CANISTER leak rate to be no greater than 10^{-4} std cm^3/sec satisfies ASME code requirements for acceptable leakage. Establishing STORAGE TUBE helium pressure within the limits of Figure 3.2-1 ensures that adequate helium remains to provide an inert atmosphere preventing the onset of corrosion

APPLICABILITY

The STORAGE TUBE receives its helium backfill and initial leak test at the conclusion of CANISTER HANDLING. Therefore, the need for the tube to maintain conclusion of CANISTER HANDLING. Therefore, the need for the tube to maintain the backfill applies only to subsequent activities, namely STORAGE OPERATIONS. Successful completion of the leak test marks the transition to STORAGE OPERATIONS and the point at which the requirements of the LCO are imposed.

ACTIONS

A Note added to the ACTIONS states that a separate Condition entry is allowed for each STORAGE TUBE containing an ISF CANISTER. Subsequent STORAGE TUBES that do not meet the LCO are governed by subsequent Condition entries and application of the associated Required Actions.

A.1

In the event that testing detects a leak in excess of 10^{-4} std cm^3/sec, the affected STORAGE TUBE is repaired or replaced to restore the leak rate to below that value within the pressure limits shown on Figure 3.2-1. Establishing these conditions ensures that leakage (and possible contaminant migration) from the tube is maintained within analyzed limits and that sufficient helium is present to maintain an inert atmosphere.

Thermal analysis shows all fuel types can be stored indefinitely in an ISF CANISTER within a STORAGE TUBE without reliance on a helium backfill for adequate heat transfer. Consequently, thermal considerations impose no time restraints on the restoration of tube leak rate.

Similarly, the material characteristics of the canister and the tube are inherently resistant to the onset of corrosion. These features, coupled with the generally dry climate of the ISF Facility, allow the interior of the STORAGE TUBE to remain in an ambient air environment for an extended period.

The helium backfill also establishes a positive pressure on the ISF CANISTER to prevent leakage from the canister to the outside atmosphere. Imposing a Completion Time of 30 days to restore the STORAGE TUBE leak rate to within limits provides a reasonable time to repair or replace the tube lid.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

A condition for entering STORAGE OPERATIONS is that the affected STORAGE TUBE has passed its pressure and interseal leak rate test. This Surveillance Requirement establishes the need to verify that the leak rate is within limits as a prerequisite to entering STORAGE OPERATIONS.

SR 3.2.1.2

To ensure that STORAGE TUBE pressure is maintained within limits throughout the storage period, this SR requires that the leak rate and pressure of 1 tube containing an ISF CANISTER in each vault be tested annually. The verification selects a representative tube for each vault to confirm its ability to maintain confinement and an inert atmosphere.

REFERENCES

ASME Boiler and Pressure Vessel Code (1998), Section V
ISF Facility Safety Analysis Report, Chapter 8

3.2 STORAGE TUBE INTEGRITY

3.2.2 Storage Vault Heat Removal System

BACKGROUND

In addition to being a secondary confinement barrier for SPENT NUCLEAR FUEL, each STORAGE TUBE provides a heat removal path for the decay heat produced by the fuel within it. Decay heat is transferred from the fuel through helium cover gas to the inner surface of the ISF CANISTER. After transferring through the canister wall, the heat moves through the helium atmosphere of the STORAGE TUBE and through the tube wall. Outside air enters the storage vault through sixteen screened inlet vents, flowing downward through fixed ducts to the lower vault area where it rises among the STORAGE TUBES. The rising air removes decay heat from the outer surface of each tube and exits through small annular openings at the surface of the charge face to leave the Storage Area through elevated outlet vents. The storage vault heat removal system is passive and depends upon natural circulation to function. An unrestricted flowpath is necessary for proper operation.

APPLICABLE SAFETY ANALYSIS

The surface temperature of the concrete vault structure is the limiting parameter for this specification. The maximum off-normal temperature of the vault concrete is limited to 200°F. With an assumed maximum normal operating fuel temperature of 118°F, a total blockage of vault airflow, and adiabatic boundary conditions, the calculated rise in the temperature of the fuel, canister, and STORAGE TUBE is 0.35°F/hr. The increase is conservatively applied to the vault's concrete surface because of its close proximity to the tube. This analysis indicates that concrete temperature reaches its limit in approximately 9½ days after airflow blockage.

LCO

Ensuring that the Storage Vault Heat Removal System is OPERABLE establishes that the stored SPENT NUCLEAR FUEL is adequately cooled to preserve its physical integrity as well as that of the ISF CANISTERS and the STORAGE TUBES that contain it.

APPLICABILITY

The Storage Vault Heat Removal System is required to operate only when SPENT NUCLEAR FUEL is placed within a STORAGE TUBE. Consequently, This LCO is applicable only during STORAGE OPERATIONS.

ACTIONS

A note added to the ACTIONS states that a separate Condition entry is allowed for each STORAGE TUBE containing an ISF CANISTER. Subsequent STORAGE TUBES that do not meet the LCO are governed by subsequent Condition entry and application of the associated Required Actions.

A.1

If the heat removal system for each STORAGE TUBE containing an ISF CANISTER is not functioning properly, action must be taken to restore cooling to maintain the integrity of the fuel and its confinement structures. Heat removal within the storage vault is driven by natural circulation, requiring a flowpath between the heat source (nuclear fuel) and the heat sink (ambient atmosphere). Blockage of this flowpath is the most likely cause for interruption of spent fuel cooling. Since calculations for a conservative adiabatic heatup show that the concrete temperature limit would not be reached for approximately 9½ days, allowing operators 48 hours to find and remedy the cause of the flow blockage is permissible.

B.1

The inlet and outlet air vents of the Storage Area can be readily cleared of obstructions if necessary. If this is insufficient to satisfy Required Action of A.1, the airflow obstruction is most likely located in the annulus between a STORAGE TUBE and the vault structure. In this case, the ISF CANISTER in the affected STORAGE TUBE must be moved to another tube with adequate cooling. Calculations for a conservative adiabatic heatup show that the concrete temperature limit would not be reached for approximately 9½ days after airflow blockage. Given this and assuming the operators used the full 48 hours allowed by Required Action A.1, an additional 96 hours is a reasonable time to relocate an ISF CANISTER into an unaffected STORAGE TUBE.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The passive nature of natural circulation cooling within the storage vault depends upon an unobstructed air path connecting the heat source (spent fuel) to the heat sink (ambient atmosphere). The purpose of this surveillance is to require a periodic inspection of locations where obstructions are most likely to occur. Given a total airflow blockage, no fuel or structural temperature limit would be approached for approximately 9½ days. With a nominal time of 48 hours between inspections, as much as 60 hours could elapse before a lack of heat removal condition is discovered if the 1.25 extension allowance of SR 3.0.2 is applied. When this is combined to the completion times of Required Actions A.1 and B.1, a total of 204 hours (8½ days) could elapse from the time of airflow obstruction until the affected ISF CANISTER is relocated into a STORAGE CANISTER with adequate cooling.

REFERENCES

ISF Facility Safety Analysis Report, Chapter 8

3.3 CRITICALITY CONTROL SYSTEM

3.3.1 Fuel Packaging Area Limits

BACKGROUND

The Fuel Packaging Area (FPA) is designed to accommodate the dry transfer of SPENT NUCLEAR FUEL from DOE-provided transport casks to ISF CANISTERS suitable for storage at the ISF Facility. Since the facility deals with three (Peach Bottom, Shippingport, and TRIGA) fuel types of differing physical and nuclear characteristics, controls are necessary to segregate these types during operations in the FPA.

APPLICABLE SAFETY ANALYSIS

Criticality analyses show that all fuel handling operations within the FPA will not result in a k_{eff} greater than 0.95. These analyses, however, assume that a single fuel type is present in the FPA at any one time. Only limited analyses have been performed to assess the criticality effects of FPA operations with a combination of fuel types. Since such operations could constitute an unanalyzed condition, administrative constraints are imposed to prevent it from occurring.

LCO

To preclude the possibility of entering an unanalyzed operating condition, the LCO restricts activities within the FPA to only one fuel type at a time.

APPLICABILITY

This LCO is applicable during LOADING OPERATIONS because this is the phase during which SPENT NUCLEAR FUEL is present in the FPA. It is also applicable during RECEIPT OPERATIONS because a Surveillance Requirement specifies the verification of the type of SPENT NUCLEAR FUEL contained in each transfer cask prior to placement in the Transfer Tunnel.

ACTIONS

The overall goal of the Required Actions is to halt fuel handling activities upon discovery of the presence of more than one fuel type within the FPA and to effect a recovery plan that restores the FPA to an analyzed condition.

A.1

When the presence of mixed fuel types is discovered in the FPA, action must be taken immediately to halt any fuel handling that could result in unanalyzed interaction between them. Immediately stopping fuel handling prevents any interaction from occurring.

A.2

With LOADING OPERATIONS suspended, no evolutions are in progress in the FPA that constitute a criticality hazard. Therefore, the facility staff can, in a deliberate manner, develop a recovery plan to restore the FPA to an analyzed condition containing only one fuel type. 72 hours is adequate to formulate this plan.

A.3

After its development, the recovery plan implements the actions needed to return the FPA to a single-fuel condition. In addition to physical activities within the FPA, the plan may require engineering evaluations, changes to procedural guidance, or other administrative details. Restoration activities could involve the return of SPENT NUCLEAR FUEL to INTEC and require several weeks to reconfigure receiving equipment. Accordingly, the Completion Time of 90 days is reasonable.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1

This SR requires verification of the type of SPENT NUCLEAR FUEL contained in the DOE transfer cask before the cask is allowed to be moved into the Transfer Tunnel. Since physical inspection of the fuel is not possible in the Receipt Area, this verification is performed by reviewing the DOE-supplied documentation accompanying the transfer cask.

SR 3.3.1.2

This SR provides visual verification that all fuel packages handled in the FPA at one time are of the same type.

REFERENCES

ISF Facility Safety Analysis Report, Section 3.3

3.3 CRITICALITY CONTROL SYSTEM

3.3.2 Criticality Monitoring System

BACKGROUND

The criticality monitoring system is a subsystem of the radiation monitoring system that employs dedicated gamma radiation detectors in the Fuel Packaging Area (FPA) to detect rapidly rising radiation levels that are consistent with the onset of criticality during the movement of SPENT NUCLEAR FUEL.

APPLICABLE SAFETY ANALYSIS

The criticality monitoring system alarm setpoints are high enough to minimize alarms from sources other than criticality and low enough to detect the minimum criticality event of concern. Setpoints for monitors are based upon modeled radiation fluxes, the position of monitors, the distance between monitors and potential sources, and the shielding effects of nearby equipment.

LCO

The criticality monitoring system must be OPERABLE as specified by the Surveillance Requirements to ensure that a critical condition does not go undetected during fuel movement in the Fuel Packaging Area.

APPLICABILITY

Because all fuel handling operations outside of the Fuel Packaging Area involve containers that prevent critical geometries, the requirement for criticality monitoring system availability is limited only to LOADING OPERATIONS when SPENT NUCLEAR FUEL is actually in the FPA.

ACTIONS

A.1

If the criticality monitoring system is not OPERABLE, LOADING OPERATIONS are immediately halted to prevent further unmonitored fuel movement.

A.2

Once LOADING OPERATIONS are suspended, no additional SPENT NUCLEAR FUEL movement is permitted until the criticality monitoring system is again OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This Surveillance Requirement ensures that the criticality monitoring system is qualitatively checked to provide reasonable assurance that is functioning properly prior to the start of fuel movement in the FPA and every 24 hours thereafter. These checks continue while SPENT NUCLEAR FUEL is present in the FPA.

SR 3.3.2.2

To provide greater assurance of proper criticality monitoring system operation beyond that afforded by the CHANNEL CHECK required by SR 3.3.2.1, this SR provides for an annual CHANNEL FUNCTIONAL TEST, the first of which to be performed within 7 days before the initial commencement of LOADING OPERATIONS.

REFERENCES

ISF Facility Safety Analysis Report, Chapter 5

3.4 FUEL PACKAGING AREA CONFINEMENT BOUNDARY

3.4.1 Heating, Ventilation, and Air Conditioning (HVAC) System

BACKGROUND

Proper operation of the HVAC system is necessary to establish airflows that prevent the spread of contamination throughout the ISF Facility as well ensure that all ventilation air leaving the facility is filtered and monitored.

APPLICABLE SAFETY ANALYSIS

In addition to establishing habitable conditions within the ISF Facility, the HVAC system limits the likelihood for contamination migration by establishing ventilation zones of different air pressures. The highest pressures are in areas least likely to become contaminated while areas with a greater potential for contamination have lower pressures. This arrangement causes air to flow from the least contaminated areas to the more contaminated ones, thereby limiting the spread of radioactive particles. The area of lowest pressure is the Fuel Packaging Area (FPA) where the potential for contamination is highest. All airflow from the FPA passes through a series of HEPA filters and is monitored for radioactivity before its release to the outside atmosphere.

LCO

Establishing that the HVAC system is OPERABLE (as defined by the accompanying Surveillance Requirements) ensures that confinement boundaries are in place and that necessary contamination filtering equipment is available.

APPLICABILITY

The generation of airborne contamination is most likely in the FPA during fuel packaging because it involves a variety of manipulations with bare nuclear fuel. Consequently, this LCO is applicable during LOADING OPERATIONS.

ACTIONS

A.1

LOADING OPERATIONS involve the movement of unshielded fuel elements within the FPA and present the most likely source of contamination. Halting these operations when the HVAC system is not available minimizes the potential for contamination production and migration.

A.2

As a further preventive measure against the spread of contamination from more heavily contaminated areas, this Required Action directs the closing of all confinement boundaries. The Completion Time of 1 hour allows this to be done in a prompt and controlled manner.

A.3

This Required Action requires local air monitoring at various points within the ISF Facility. This ensures that facility personnel are aware of areas having airborne contamination levels in excess of permissible limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR requires verification that 1 FPA exhaust fan is running prior to and during LOADING OPERATIONS. This ensures that the FPA is maintained at a negative pressure to prevent the spread of contamination. It also ensures that airflow from the FPA passes through HEPA filters to remove particulates and is monitored prior to release to the environment. The requirement to verify fan operation every 24 hours ensures that operators are aware of HVAC system status and serves as a backup to installed instrumentation and alarms.

SR 3.4.1.2

A critical function of the HVAC system is the ability to filter radioactive particulates before they enter the environment. An excessively low differential pressure across the HEPA filter bank indicates the presence of a punctured filter. An excessively high differential pressure indicates a blocked filter. This SR requires verification that the filter bank differential pressure is within a range that shows all filters are functioning effectively.

SR 3.4.1.3

This SR establishes requirements for the condition of major penetrations into the FPA to ensure that confinement is established whenever SPENT NUCLEAR FUEL is present.

When fuel enters the FPA for packaging, it moves through the normally closed cask port separating the FPA from the Transfer Tunnel. With the port plug removed, an inflatable seal closes the gap between the cask and the port and serves as a confinement boundary to prevent the spread of contamination. The seal also maintains proper ventilation flow by preventing direct airflow from the Transfer Tunnel to the FPA, thereby bypassing the FPA inlet HEPA filter.

Packaged fuel exits the FPA through the normally closed canister port and into the canister trolley for transfer to the Canister Closure Area. An inflatable seal, similar to the one at the cask port, closes the gap between the canister and the canister port, providing the same confinement and ventilation control as the cask port seal.

Neither waste port has an inflatable seal comparable to those associated with the cask and canister ports. Consequently, the SR prohibits opening either of ports while LOADING OPERATIONS are in progress unless SPENT NUCLEAR FUEL in the FPA is in designated storage locations. Opening a waste port provides direct communication between the FPA and the Solid Waste Storage Area that is not within the confinement boundary. The risk of contamination migration is minimal, however, because the HVAC continues to maintain the FPA under a negative pressure and no fuel handling is in progress.

REFERENCES

ISF Facility Safety Analysis Report, Section 4.3.1